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Safety assessment of proposed improvements to RBMK nuclear power plants

Report of the IAEA Extrabudgetary Programme on the Safety of RBMK Nuclear Power Plants

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SAFETY ASSESSMENT OF PROPOSED IMPROVEMENTS
TO RBMK NUCLEAR POWER PLANTS
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FOREWORD
by the Director General

The IAEA initiated in 1990 a programme to assist the countries of eastern Europe and the former Soviet Union in evaluating the safety of their first generation WWER-440/230 nuclear power plants. The main objectives of the Programme were: to identify major design and operational safety issues; to establish international consensus on priorities for safety improvements; and to provide assistance in their implementation.

The scope of the Programme was extended in 1992 to include RBMK and WWER-1000 plants in operation and under construction. The Programme complements ongoing IAEA activities on WWER-440/213 plants.

The Programnme is pursued by means of plant specific safety review missions to assess the adequacy of design and operational practices; Assessment of Safety Significant Events Team (ASSET) reviews of operational performance; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Other components are: follow-up safety missions to nuclear plants to check the status of implementation of IAEA recommendations; assessments of all safety improvements implemented or proposed; peer reviews of safety studies; and training workshops. The IAEA is also maintaining a database on the technical safety issues identified for each plant and the status of implementation of safety improvements. An additional important element is the provision of assistance by the IAEA to strengthen regulatory authorities.

The Programme is extrabudgetary and depends on voluntary contributions from IAEA Member States. Steering Committees provide co-ordination and guidance to the IAEA on technical matters and serve as forums for exchange of information with the CEC and with other international and financial organizations.

The Programme, which takes into account the results of other relevant national, bilateral and multilateral activities, will provide a technical basis for safety related decisions to be made by the countries operating WWER and RBMK plants and by countries providing technical and financial support for upgrading the safety of nuclear power plants in these countries.

The IAEA further provides technical advice in the co-ordination structure established by the Group of 24 OECD countries through the Commission of the European Council to provide technical assistance on nuclear safety matters to the countries of eastern Europe and the former Soviet Union.

PREFACE

Since the accident at Chernobyl Unit 4 in 1986, RBMK specialists have identified a number of modifications made to enhance the safety of RBMK reactors. Some of these modifications have already been made, while others are still in the implementation or planning stages. As a result of work which was started in 1986, other potential modifications have been identified for further consideration.

The purpose of this report is to summarize the findings and recommendations of a Consultants Meeting convened by the IAEA in Vienna (27 October - 5 November 1992) to review new design features and modifications proposed or already implemented for RBMK reactors. This information was provided in four technical areas, namely: Core Monitoring and Control, Pressure Boundary Integrity, Accident Mitigation and Electric Power Supply.

The report also presents the status of the modifications at the plants as given by the RBMK specialists.

The limited information available and the time constraints did not allow the review to be conducted at the level of a peer review, and the findings and recommendations made reflect the limited scope of the review. More detailed reviews and analyses focusing on selected safety issues are required and should be conducted on a generic and plant specific basis as appropriate.

In Chapters 2-5 of the report the main findings and recommendations for the four topical areas reviewed are summarized. Appendices I-IV reflect the results of the discussions held at the meeting and provide more detailed information on the review.

CONTENTS

1. INTRODUCTION

Following the Technical Committee Meeting convened by the IAEA in April 1992, the IAEA announced, in June 1992, an Extrabudgetary Programme on the Safety of RBMK Nuclear Power Plants and invited its Member States to assist in its implementation. Along the lines recommended by the Technical Committee a work plan has been developed for the programme implementation which is focusing on:

- assisting in the definition of scope and priorities for technical work required;
- organizing specialists meetings on topical areas of RBMK safety;
- conducting plant specific safety missions to relevant plant sites to review specific topics related to design and to the conduct of operation;
- serving as a forum to establish international consensus on safety improvements required;
- peer review results;
- providing training.

In addition to the improvements in areas important to safety which are to be implemented immediately a most important task is a comprehensive safety review of RBMKs.

As a first step in the programme implementation, a Consultants Meeting to assess the safety improvements already proposed was organized by the IAEA in Vienna from 27 October - 5 November 1992 on the Safety Assessment of Proposed Improvements of RBMK NPPs.

The objective of the meeting was to review the scope and results of safety evaluations performed and the technical basis for the safety improvements implemented and planned for the RBMK reactors. Areas where further work is needed were specified and shortcomings of work performed or planned were identified. The results of this meeting will contribute to establishing a basis for further activities and decisions on technical and financial support for implementation of measures to improve safety.

Based on the papers presented during the Technical Committee Meeting on Safety of RBMK Reactors, and the discussions during that meeting, four technical areas were selected for review:

- Core Monitoring and Control;
- Pressure Boundary Integrity;
- Accident Mitigation;
- Electric Power Supply.

In order to facilitate the review, supporting documentation was prepared by RBMK specialists, in English, in each of the above areas of review [1-17]. In addition, further background information was provided by the RBMK specialists participating in the meeting.

The information available for review included, in general, design objectives and concept, system description, including system operation and interaction, safety concerns already identified by designers/operators, proposed safety improvements, including applicability and status of implementation to each individual RBMK unit and information on differences/modifications of the existing three generations of RBMKs and on plant specific status.

A commonly accepted strategy for the evaluation of nuclear power plant safety is based on the defence in depth concept. For this reason, the review carried out was to the extent possible structured and performed in terms of the performance and integrity of basic safety functions and the protection of the physical barriers.

TABLE I. MAJOR SAFETY RELATED MODIFICATIONS OF RBMK NUCLEAR POWER PLANTS: STATUS BEFORE AND AFTER RECONSTRUCTION

Modification	Kursk 1		Ignalina 2		Smolensk 3		Objective
	Before	After	Before	After	Before	After	
1. EMERGENCY CORE COOLING SYSTEM							Reliability and scope of ECCS upgrade
Cooling pumps for damaged part of reactor		3	6		6		
Cooling pumps for undamaged part of reactor		3			3		
Accumulators (water volume, 1 accumulator)	1 x 6 (6.3m^3)	3 x 6 (12.5m^3)	2 x 8 (12.5m^3)		2 x 6 (12.5m^3)		
Distribution group headers check valves		2 x 22	2 x 22		2 x 22		
Emergency feed pumps	3	5 (min)	6		3		
Headers and pipes of ECCS	2 x 3	2 x 3	2 x 3		2 x 3		
2. PRESSURE BOUNDARY INTEGRITY MONITORING/ CONTROL							Early leak detection, prevention of brittle failure
Automated NDE		+		+		+	
Leak detection (humidity, radioactivity, acoustic)		+		+		+	(System tested at Leningrad 1)
3. OVERPRESSURE PROTECTION SYSTEM (max. number of failed fuel channels)							Reactor cavity overpressure protection for simultaneous failure of given number of fuel channels
Present status	3		3		9		
After I. reconstruction stage, 2 safety valves, 2 pipes ∅ 400 mm		5					
After II. reconstruction stage, 2 pipes ∅ 600 mm (need to cut biological shield), redundant support system		9		9			
4. RADIOACTIVE RELEASE PROTECTION SYSTEM	Condenser	+	ALS (Bubbler)	NN	ALS (Bubbler)	NN	Reduction of radioactivity release to environment

TABLE I. (cont.)

Modification	Kursk 1		Ignalina 2		Smolensk 3		Objective
	Before	After	Before	After	Before	After	
5. BUILDING FOR RELIABLE POWER SUPPLY							
Remote shutdown panel		+	+		+		
Diesel generators	3 x 3,5 min.	+3 x 6,2	6 x 5,6		3 x 6,2 2 x 1,6		
Independent power supply trains	1	3	6		3		
6. FIRE PROTECTION UPGRADES		+		+		+	
7. WATER SUPPLY FROM EXTERNAL SOURCES (FOR SEVERE ACCIDENTS)							Severe accidents mitigation
Service water	+						
Underground water (artesian)	+						
Independent line from reliable source (lake)		+		+		+	
8. STRUCTURAL UPGRADES							Under discussion
Reactor hall structural and leak tightness upgrade							
9. CPS AND I & C UPGRADES		+	NN	NN		+	
10. SEISMIC RESISTANCE IMPROVEMENT		+		+		+	
11. ADDITIONAL SCRAM SIGNAL							
Shutdown margin		+		+		+	
Low flow on DGH		+		+		+	
Pressure drop in main circulation circuit		+		+		+	
12. AUTOMATED ON-SITE RADIATION MONITORING		+		+		+	
13. GENERATOR BREAKERS		+	+		+		
14. DISCONNECTION OF GENERATOR FROM GRID FROM CONTROL ROOM		+		+		+	Due to fire at Chernobyl 2
15. UPGRADING OF DRAIN SYSTEM		AT	NN	NN	NN	NN	Prevent drain overflow condensation
16. ECCS UPGRADE - ADDITIONAL WATER TANK		+	NN	NN	NN	NN	

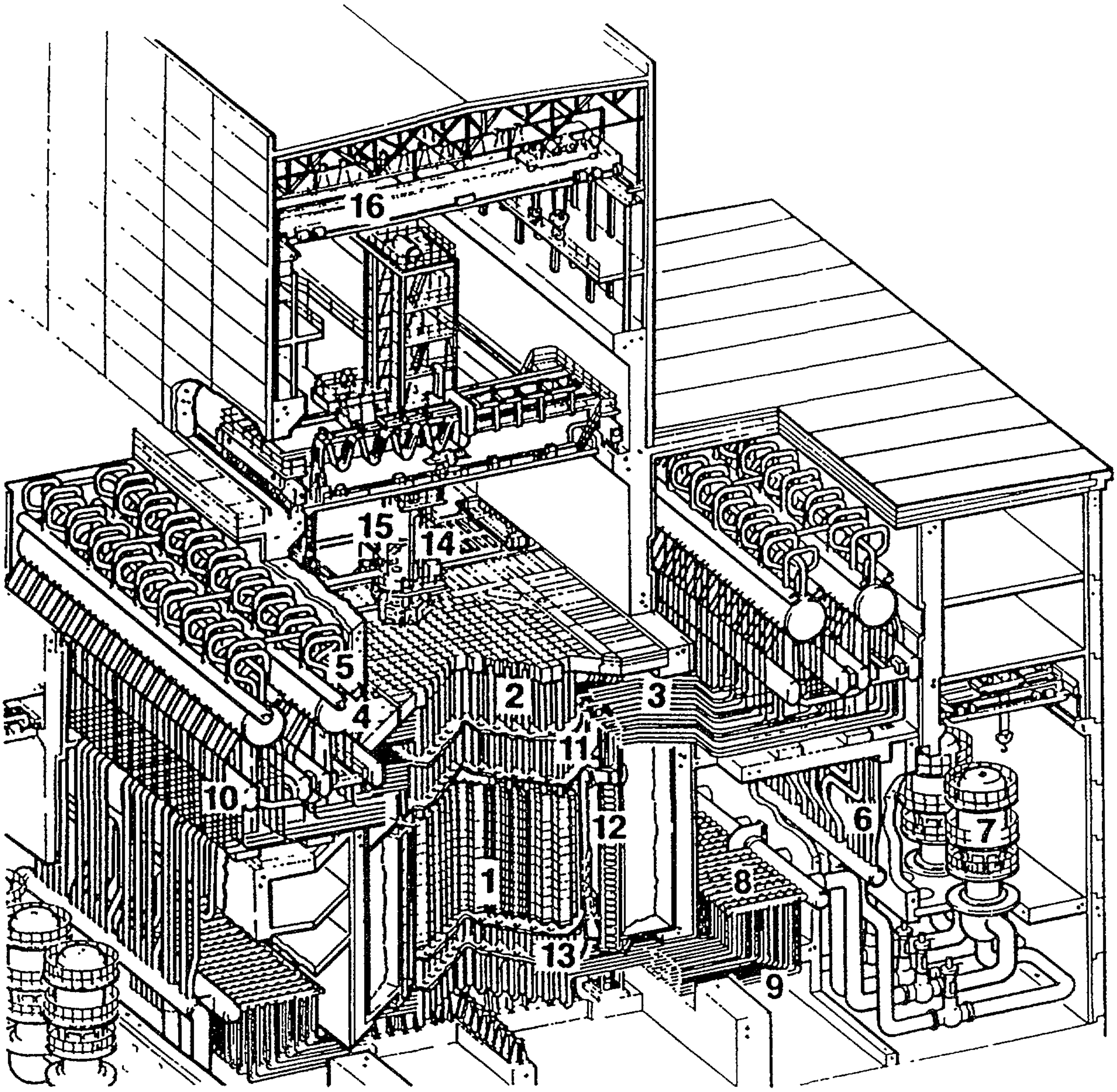
Note: AT = additional tanks
NN = not needed
\+ = implemented

TABLE II. PLANT SPECIFIC STATUS

Unit	Generation	Status	Number of CPS Channels	Number of Fuel Channels
Chernobyl 1	1	operational	179	1693
Chernobyl 2	1	shutdown	179	1693
Chernobyl 3	2	operational	211	1661
Kursk 1	1	operational	179	1693
Kursk 2	1	operational	179	1693
Kursk 3	2	operational	211	1661
Kursk 4	2	operational	211	1661
Kursk 5	3	constructed	223	-
Leningrad 1	1	operational	191	1693
Leningrad 2	1	operational	179 (191)	1693
Leningrad 3	2	operational	211	1661
Leningrad 4	2	operational	211	1661
Smolensk 1	2	operational	211	1661
Smolensk 2	2	operational	211	1661
Smolensk 3	3	operational	211	1661
Ignalina 1	2	operational	211	1661
Ignalina 2	2	operational	211	1661

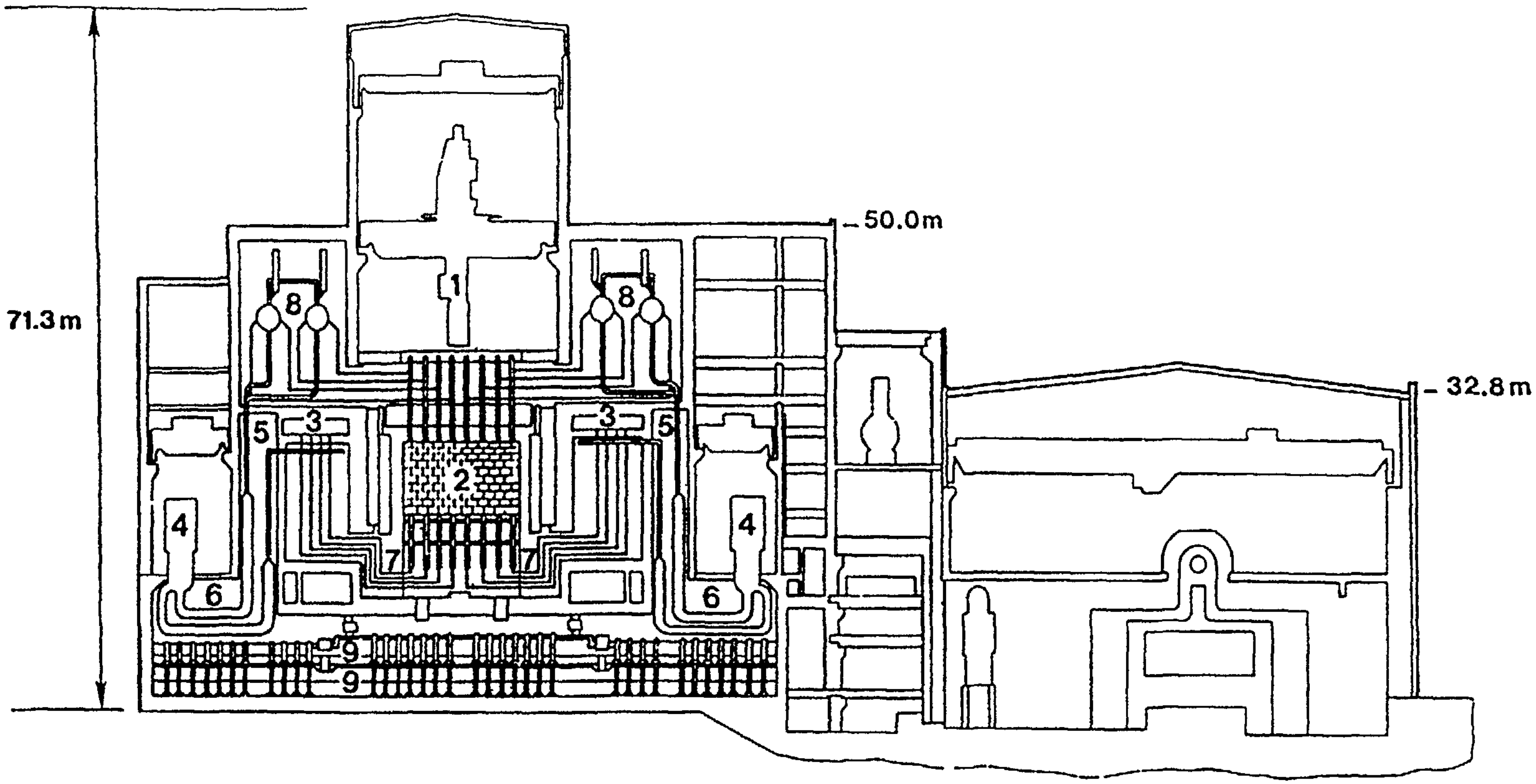
The review was conducted addressing the technical areas mentioned in four parallel working groups. The meeting was attended by 28 safety experts from 13 countries and the CEC and 16 RBMK specialists from Russia, the Ukraine and Lithuania.

The following chapters present the findings and recommendations for the four technical areas reviewed. Appendices I-IV reflect the results of daily discussions and provide more detailed information on the reviews performed. In Tables I and II the list of modifications summarized by RBMK specialists is given, along with selected plant specific characteristic features. Figures 1, 2, and 3 show the RBMK plant cross-section, elevation and flow diagram.



1 Reactor	**9 Reactor inlet water pipes**
2 Fuel channel upper tract	**10 Burst-can detection system**
3 Steam/water pipes	**11 Upper biological shield**
4 Steam separators	**12 Side biological shield**
5 Steam headers	**13 Lower biological shield**
6 Downcomers	**14 Irradiated fuel storage pond**
7 Main circulating pumps (MCP)	**15 Re-fuelling machine**
8 Group distribution headers	**16 Bridge crane**

FIG. 1. Sectional view of RBMK 1000 nuclear power plant.



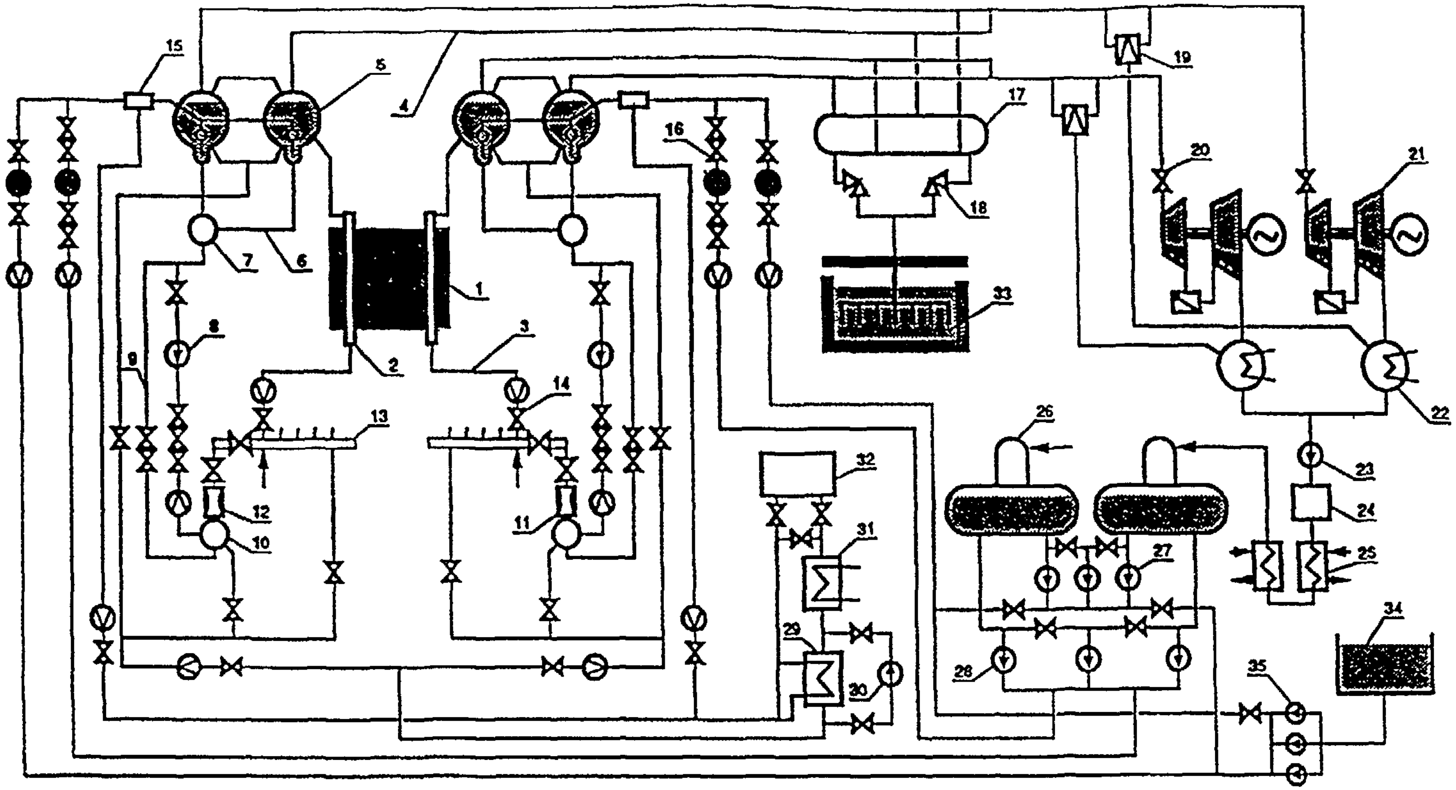
Reactor Hall

1 Re-fuelling machine
2 Reactor
3 Pressure header
4 Main circulating pumps
5 Downcomer pipes

Turbine Hall

6 Suction header
7 Reactor inlet water pipes
8 Steam drums
9 ALS pools

FIG. 2. Elevation of RBMK 1000 nuclear power plant.



1 Reactor
2 Fuel channel
3 Water pipelines
4 Steam pipelines
5 Steam separator
6 Downcomer
7 MCP suction header
8 Main circulation pump
9 MCP header bypass
10 MCP pressure header
11 Mechanical filter
12 Flow limiter
13 Group distribution header
14 Isolation and control valve
15 Mixer
16 Feedwater valve assembly
17 Steam headers
18 Main relief valve (MRV)
19 Steam dump valve
20 Turbine trip valve
21 Turbogenerator
22 Condenser
23 Condensate pump
24 Condensate purification
25 Heater
26 Deaerator
27 Auxiliary feedwater pump
28 Feedwater pump
29 Blowdown regenerator
30 Cooldown pump
31 Blowdown afterheat
32 Bypass purification
33 ALS pool
34 Emergency water tank
35 Emergency feedwater pump

FIG. 3. Flow diagram of RBMK 1000 nuclear power plant.

2. CORE MONITORING AND CONTROL

The main changes since the Chernobyl accident are related to the protection functions based upon neutron flux measurement. This appears to be because the Chernobyl accident was a reactivity induced accident, which has influenced much of the backfitting work. There have been several modifications implemented which have resulted in a reduction of the positive void coefficient and mitigation of its effect under accident conditions. After the introduction of the "urgent measures" to improve safety, several longer term items were identified. These additional measures are currently in various stages of implementation. Review of the information presented at this meeting indicated that the following items are still in progress. Some future modifications were also identified during the meeting. The status of many of these modifications is summarized in Tables VI and VII of Appendix I to this report.

2.1. DESIGN MODIFICATIONS

Implemented

- introduction of approximately 80 additional fixed absorbers (52 for Ignalina NPP);
- increased operational reactivity margin (ORM) and increased minimum ORM;
- introduction of 21 to 24 fast acting control rods, which constitute the fast scram system (FSS) with an insertion time of 2.5 s;
- increase in the number of bottom inserted rods, all of which are now inserted when the scram occurs;
- modified control rod design to eliminate the positive scram which contributed to the Chernobyl accident;
- increase in speed of insertion of control rods from about 19 s to 12 s;
- ORM calculation is made every 5 minutes, instead of every 15 minutes;
- improvements in the computational capabilities of core physics codes;
- strengthening of administrative procedures.

Under way

- increase in fuel enrichment from 2.0% to 2.4% (currently being introduced on all plants except Ignalina);
- installation of the modified control rods for first generation plants;
- "modernization" of the control and instrumentation system.

Planned

- an additional fast acting shutdown system potentially using ^3He injection;
- further enhancement of the control and protection system (CPS) cooling system to reduce the maximum reactivity insertion resulting from system failure;
- new scram signals based on reactivity margin;
- graphite stack mass reduction by 20% for Kursk 5.

The effect of these modifications has been reviewed and results in the following findings.

2.2. FINDINGS

- All of the fixes which have been made to the CPS and ORM appear to be good and should result an the improvement of the safety of RBMK reactors. Overall, these design changes should increase the safety of the reactors in many respects, but it was not evident to the

western experts that the proposed changes will not reduce the safety margins in other respects. The material presented does not include all the details of the analyses used to make this trade-off between increased safety and possible reduced safety margins. The proper assessment of the real trade-off of the competing and interacting design fixes requires three-dimensional analyses.

- There is a plan to increase the number of redesigned control rods in the future. The redesigned control rods are also to be implemented at the RBMK second generation plants.

- With the addition of the 80 permanent absorber rods, using conservative initial conditions such as minimum ORM of 30 rods, fuel enrichment of 2.0%, shortest channel voiding time, etc., the analysis shows that the resulting void reactivity effect is 1.7ß. With an ORM of 45 rods, the effect is 1.2ß. The actual effect has been measured as 0.8ß with the 2.4% enriched fuel, which has been added to date. This latter value is within the capability of the shutdown system and is a very significant reduction from the value prior to the backfitting measures being implemented.

- Neutronic calculation capability has been improved by the introduction of codes such as WIMS D4 and MCU. These codes have been used to improve data and validate existing Russian codes by comparison exercises. Developments are under way to make further improvements by the introduction of more 3-D codes for both dynamic and fuel management calculations.

- In general it appeared that the monitoring and protection systems were much better organized and more comprehensive in scope than was apparent from the material prepared for the review.

- The monitoring of plant parameters and processing of plant information were found to be extensive. The SKALA and TITAN computer systems would appear to be overloaded and currently have a slow cycle time, e.g. update of flow every 40 seconds and dryout margin every 5 to 15 minutes. The efforts and actions to accelerate the rate of data processing and information updating are supported by the experts.

- A complicated combination of manual control and automatic control is utilized to maintain total reactor power, as well as the power distribution. The SKALA central monitoring system, as well as other instruments, provide the operator with information concerning the power level and power distribution. The SKALA system, which provides updated information every 5 minutes, is designed to provide guidance to the operator for steady state control of the power distribution. There is a very heavy reliance by the operator on accurate information being presented by the SKALA system.

- The SKALA system also calculates the ORM. The ORM is expressed in terms of the number of "equivalent" control rods remaining within the core. This definition is not precise, but the ORM is very important to safety because of its effect on the void and power coefficients. It was also noted that when a violation of the ORM occurs, the operators are only required to scram by following an administrative procedure, since this has not been implemented into an automatic engineered safety feature.

- There is a heavy dependence on operator actions to ensure plant safety.

- The procedures and practices for refuelling the reactor, as described in the material presented, are comparable to those used for on-power refuelling in Canada.

- The change of manual shutdown button, from one that had to be manually held down until the plant was shut down, to one that need only be pressed once is most justified. The actions to

reduce the time for rod insertion and to enhance the speed with which the reactor is shutdown are supported by the experts.

- The shutdown algorithms require further study by western experts and the links to the accident transients and to the plant safety case must still be made. The approach to proof and periodic testing as well as the role of such testing when making the high claims for shutdown system reliability should be discussed additionally.

- The division of the control rods of the same function among the six power supplies was considered as good practice, but there remains the question of the extent of segregation of the cable runs and the location of processing equipment.

- Calculations of the power pulse resulting from channel voiding take into account electronic delays (0.1 s) and actual rod drop times (2 to 2.5 s) as measured both experimentally and at NPPs. The electronic delay time seems small.

- In pipe break scenarios, all reactor power levels have been considered, even below 20% where power operation is forbidden, and the results presented show power pulses within acceptable ranges.

- Rod withdrawal accidents cause a peak in the neutron flux in the vicinity of the withdrawn rod. If this type of accident is analysed assuming the control and protection system works, then the flux distribution changes due to insertion of other rods in the vicinity. Depending on the specific conditions, the flux distribution can actuate either the local control system or cause the power scram system (PSS) to activate. The effectiveness of the response of these systems depends upon whether there is a 7, 9 or 12 zone control system and where the location of the rod is relative to the detectors. In the case of Kursk, there is only an out of core control system so coverage will be poorest in that reactor.

- The power peaks resulting from rod withdrawal are calculated using very conservative starting assumptions. The results presented show that for the majority of cases considered, the targets of maintaining the cladding temperature below 1200°C and preventing fuel melting are achieved. Only in the case where all conservatisms are used, will very limited fuel melting occur in one channel for those reactors with 8 zone control systems. Therefore, for those reactors, an administrative procedure has been instituted which requires that the drive mechanisms be electrically disconnected when some of the central rods are fully inserted. This procedure is followed at Kursk NPP.

- Protection against loss of water from the CPS cooling circuit is covered by 3 parameters, related to pressure in the header, water flow rate and level of water in the upper tank, which initiates water makeup to systems with breaks. The specific rate of water loss depends upon the design of CPS rods and their position in the core. The control rod design change has reduced the positive reactivity insertion from about 4ß to 5ß to about 2.5ß. Even though this is close to the capability of the FSS the results of the analysis presented show that maximum rate of reactivity addition is 0.125 ß/s. Obtaining of acceptable results from the analysis of this event is clearly dependent on the accuracy of the codes used to predict the rate of discharge of water from the channels. This needs to be confirmed by further experiments, or ways of physically reducing the rate of reactivity addition need to be found.

- There was no discussion of accidents involving faulty ECCS initiation, gas ingress from the ECCS accumulators to the core and group rod withdrawal, although it was stated that the group rod case would be easier to detect and compensate for because of the larger perturbation.

- The number of channels with 2.4% fuel enrichment varies from unit to unit as summarized in Table VII. There is no plan to load 2.4% enriched fuel into the Ignalina cores.

- Modified control rods are being installed on all first generation units (Table VII). The number of bottom entry rods has not been increased to 32 on first generation units, because it gave rise to problems in maintaining the reactor subcritical on shutdown.

- A modernized version of the control and instrumentation system has been developed and should be installed starting in early 1993. This new system has increased diversity and segregation of functions and a more rapid cycling of the data acquisition system. Additional protective functions are contained in the new system, for example actuation of the FSS in response to signals from in-core detectors.

From the limited discussion of the details of these modifications during this meeting, all were perceived to be positive steps towards enhancing the safety of RBMK reactors.

2.3. RECOMMENDATIONS

In the light of the discussions concerning all of the above modifications to systems and analytical calculations the following recommendations were made:

- Selected information from the extensive references quoted in the supporting documentation should be made available for independent review to enhance western experts' confidence in the significant improvements reported at this meeting. This review will require extensive and detailed discussions.

- Efforts to improve 3D-methods for neutronic calculations and static and dynamic analysis of power distribution should be continued and extended to cover fuel management. A common approach agreed by RBMK and western specialists should be used.

- Further review should be undertaken by Russian authorities to minimize the high reliance of safety functions on operator action. An increase in the scope of automatic functions may be warranted.

- The RBMK designers should be encouraged to introduce additional instrumentation and control devices, where such additions can be shown to improve safety.

- Further studies should be undertaken to remove the necessity to maintain sufficient ORM in order to have an acceptable void coefficient.

- The replacement of the ORM concept, which requires too much operator involvement, should be further explored.

- Additional information on the control and protection strategy adopted for RBMK plants should be provided. In particular it is necessary to identify the design requirements of the safety systems and clearly link them to the safety case.

- The implications of the change in local power control system, e.g. with regard to change in detector type and introduction of 12 zone controllers, should be peer reviewed.

- The data processing computers should be upgraded and better alarms and displays installed.

- Integrated reviews of both the neutronic and thermal hydraulic studies are needed since there is significant interaction between them.

- Independent calculations should be undertaken, using typical western safety analysis assumptions, to assess the margins which would normally be expected when using such

assumptions. Comparative analysis of RBMKs and relevant western plants should be carried out, based on results of independent calculations.

- Further experiments are needed to understand the blowdown characteristics of the main circulation circuit and of the whole heat transport system and their effect on reactivity addition.

- Procedures for experimental measurement of important plant safety parameters, e.g. void reactivity coefficient, should be reviewed.

- The modernization of the CPS should proceed expeditiously since the reported improvements would provide considerably better coverage of rod withdrawal accidents and for single channel failures when fast response detectors are installed.

- The effectiveness of the fast acting shutdown system should be increased to enhance the safety margins for reactivity induced accidents. An optimization study is necessary.

- The completeness and appropriateness of the safety criteria being applied to the RBMK should be reviewed.

No attempt has been made to rank the order of the recommendations. It was deemed not possible to make such a ranking as the experts did not have the comprehensive overviews of the plant and its safety which would be required to prioritize the recommendations.

3. PRESSURE BOUNDARY INTEGRITY

The subject of the pressure boundary was reviewed on the basis of the four subject areas, Design of Fuel Channels, Reactor Coolant System Design, Reactor Pressure Boundary Inspection and Leak Before Break Concept. The Reactor Coolant System Design was further divided into the primary and the steam circuits. The complete circuit design, as well as the modification status, was covered on the basis of the information presented verbally and in writing by the RBMK specialists. The working atmosphere was cordial, co-operative and very open, and the limits in information were governed by the limits to the knowledge of the participants.

The recommendations on the system are mainly in the areas of additional analysis, verification inspections of the systems, and peer reviews on analyses already carried out. Recommendations were also made in an attempt to improve containment confinement and to consider automatic regulation of valves.

The recommendations on improvements were in the areas of peer review of the improvements, of continuing to work on the analysis of the leak before break concept, and of high reliability leak detection systems if leak before break is to be used for reactor safety justifications.

3.1. DESIGN MODIFICATIONS

Several modifications have been implemented to enhance the pressure boundary integrity at RBMK nuclear power stations; some are under way and others are being studied.

Implemented

- Some 46 fuel channel tubes have leaked in the dissimilar weld region and have subsequently been replaced. Leakage was from stress corrosion cracks in the stainless steel section of this weld, associated with low titanium concentrations.
- Delayed hydride cracks (DHC) have been observed on the outside diameter of fuel channel tubes. Growth of DHCs has been associated with high residual stresses, induced by the former method of straightening tubes. The method used at present has overcome this problem (reduction of residual stresses).
- Some steam/water pipes supports have been added to deal with the situation observed related mainly to piping vibration.
- In the past an acid medium was used for decontamination of the circuit but was found to corrode the carbon steel component sections and the practice has been discontinued.
- A simplified engineering approach, the strain intensity factor concept and limit stress approach have been used to develop tables of allowable flaw sizes, which regulate which flaws must be repaired upon discovery during in-service inspection.
- Probabilistic fracture mechanics calculations have been performed for the 800 mm pipe to predict leak and rupture probabilities. Failure of this piping has the severest consequences should a large break occur.

Under way

- A new design of flow control valve is being installed at all plants, as a measure to prevent ruptures of fuel channels that have been caused by lack of coolant flow when valves have malfunctioned.

- A sample of fuel channels is now being inspected at all plants by ultrasonic and eddy current methods; these inspections are in addition to continuing hot cell examinations of fuel channels which are being removed for purposes of material properties studies (surveillance samples manufactured of fuel channel metal are also studied).
- Fuel channels are being (or will be) replaced at all plants with new channels (of essentially the same design) after service exposure (creep) has resulted in closure of the gap between the channel tube and graphite. This extends the service life under the same operating conditions and to a limited extent improves safety due to enhanced pre-service inspection capabilities.
- Seismic analyses of the pressure boundary are being performed for all plants on a site specific basis to resolve concerns with previous generic evaluations of seismic effects.
- Automated equipment has been developed for inspection of the main circulation circuit welds. This equipment with imaging techniques for data display is being used on a pilot basis at the Leningrad plant before being implemented at all RBMK units.
- The leak before break (LBB) was not the technical basis for the RBMK safety concept. However, analyses of LBB are in a state of development and will be used to properly distribute resources (e.g. leak detection versus ultrasonic detection of cracks).
- Evaluations of severe wind loads are performed on a plant specific basis.

Planned

- A new regulatory document on fuel channel inspection is being prepared.
- Tests of the effects of tube burst on adjacent tubes have been proposed. A 5 x 5 full-scale mockup of cells has been constructed at a facility in the Ukraine, but the tests have been delayed following the dissolution of the Soviet Union.

3.2. FINDINGS

- Only three incidents of ruptured channels have occurred during operation but no beyond design basis accidents have occurred to date (no leaks in adjacent channels). There have been 73 cases of detectable leakage resulting from through wall cracks smaller than critical size.

- Operation of the refuelling machine was also reviewed in detail including the methods of sealing both the channel closure and the refuelling machine. Seismic qualification of channels and the refuelling machine is of concern.

- The main circulation circuit is mainly of stainless steel, although there are some carbon steel valves; the second circuit is mainly carbon steel. Leak detection has been used to monitor both fuel channels and the primary coolant circuit, and the detection methods have been effective in finding leaks in degraded RBMK components.

- Non-destructive examinations (NDE) of fuel channels by ultrasonic testing (UT) or other methods is only now being implemented at all plants, and will only inspect a small sample of channels when implemented. Leak monitoring will remain the main method of detecting degraded tubes.

- Leak before break evaluations have been performed on a generic basis, with inspection data from Ignalina used to support evaluations. No modifications to plants have yet been made or proposed based on LBB evaluations.

- Continued evaluations of the leak before break concept are being performed and the results of these studies will help define the requirements for improved systems for leak monitoring.

3.3. RECOMMENDATIONS

- Volumetric 100% inspections of the fuel channels should be considered for implementation to analyse the development of sub-critical cracks. An eddy current type of device carried by the fuelling machine may enable this inspection to be done with each fuel change.

- Precise methods of prediction of the gap between the fuel channel and the graphite stack should be developed on the basis of graphite and tube behaviour.

- Fail safe characteristics of replacement flow control valves should be fully verified. Modifications should be considered to allow non-operator-initiated response to flow restrictions.

- The inspection programme should be optimized by identifying the most risk significant location. This could lead to more effective inspection programs with fewer inspections, but of a higher level of reliability. Activities already under way in this direction should be encouraged.

- The possibility of backfitting better containment-confinement systems should be studied.

- The site specific seismic analyses should be peer reviewed, and any weaknesses indicated in equipment or support should be corrected and improvements backfitted. Of special interest is the support of the refuelling machine, which should be analysed to maintain a condition of remaining locked on to a channel during a seismic event without leakage or any damage to the channel or the machine.

- The joint and nozzle connection of the feeder tubes (diameter 68 mm) to the steam separators should be further studied for the effects of ageing.

- New methods of decontamination of the circuit should be developed to reduce radiation exposure of the in-service inspection personnel without eroding the material of the circuit.

- The effectiveness of NDE methods should be more extensively established through performance demonstrations on specimens with realistic service type defects.

- The inspection of the main circulation circuit should be carried out regularly (100% UT every 4 years). Steam pipes between the separators and the turbine stop valves, and the feedwater lines between the separator and the feedwater control valves should also be inspected sufficiently.

- Currently available results related to application of the LBB concept do not provide a basis for decisions on RBMK safety.

- Development of the analysis method for pipe rupture, including zirconium, stainless steel, and carbon steel materials and taking into account earthquakes, should be continued and the probability analyses should be conducted for each specific equipment layout.

- Providing that the LBB concept is to be applied, plant specific work completed and planned should be analysed and peer reviewed, including stress analysis, fracture analysis, stability analysis, integrity analysis and leak detection methods.

4. ACCIDENT MITIGATION

In the area of Accident Mitigation the review was split into four parts covering Emergency Core Cooling System, Loss of Coolant Accident Analysis, Confinement and Reactor Cavity Overpressure Protection System, covered in Sections 4.1 to 4.4.

4.1. EMERGENCY CORE COOLING SYSTEM

The focus of the discussions was on understanding the basic design and function of the emergency core cooling system (ECCS) for the various RBMK units. Three reference plants Kursk 1, Ignalina 2 and Smolensk 3 were addressed; however, there were some limited discussions on other units (especially Leningrad 1 and Chernobyl 1, since the safety improvements have been implemented mainly to these units).

The specific upgrades or modifications that have been proposed by the RBMK specialists were also discussed. In general these upgrades are intended to improve the ECCS for the 1st generation of RBMKs to be comparable with the 2nd and 3rd generations.

In general it seems that all of the proposed modifications are safety improvements and will enhance the capacity and redundancy of the ECCS for the 1st generation RBMK units. However, the review was of insufficient detail to assess whether or not these upgrades fully address all the demands placed on the ECCS for the full spectrum of accident scenarios (1st generation RBMK).

4.1.1. Design modifications

The specific modifications addressed during our meeting are shown in Table I. Those relating to improving the ECCS are items 1, 7, 11, 15, and 16 and are described briefly below:

Item 1 - ECCS improvements

This item includes several different improvements as follows:

- Increasing the number of EFPs from 3 to 5 and the number of ECCS lines from 1 to 2 (1st stage).
- Installing additional ECCS pumps (3 for damaged core side cooling and 3 for undamaged core side cooling) and the associated 3 divisions of piping (2nd stage).
- Installation of check valves between the distribution group headers (DGH) and the main coolant pump discharge header.
- Installing additional, larger capacity accumulators.

Item 7 - Additional water sources

In Kursk 1 NPP the capability to make use of diverse external water sources for ECC injection has already been implemented (service water connection, artesian wells suction). Further improvements have been planned for all the considered plants in order to make emergency water supply from external sources (i.e. lakes) available without any need for energy from in-plant energy sources.

The main purpose of the above modification is to enlarge the spectrum of beyond design basis accidents the plants will withstand.

Item 11 - Safety system actuation

Additional scram signals are planned to be included in the reactor protection logic in order to introduce new automatic trip signals and to diversify physical parameters able to initiate scram in some critical scenarios.

During the discussion of ECCS initiation logic, a similar improvement has been found to be effective for these logic.

Some ECCS initiation logic depend on the monitoring of a single physical parameter (e.g. drum separator level), and for some events both the scram and the ECCS rely on a single physical parameter (e.g. pressure increase in a compartment room).

Item 15 - Drain system upgrading

This concerns the installation of an additional tank to store the water drained from the lower room located below the reactor cavity. This tank is to be added to the plants of the 1st generation. Its main purpose is to prevent flooding in this area.

Item 16 - ECCS water supplies

Incorporation of an additional source of water of about 2000 m^3 from where the ECCS pump could take suction. This will ensure the availability of cooling water for a longer period of time. This modification is planned for the first generation plants.

4.1.2. Findings

- Regarding the specific upgrades discussed, it was agreed that all the upgrades are safety improvements and will enhance the capacity and redundancy of the ECCS for the 1st generation RBMK units.

- It was also found that there has been insufficient information presented to assess whether or not these modifications will fully address the demands placed on the ECCS for a full spectrum of accident scenarios, including those intended to be covered by the planned improvements.

- It should also be noted that there has been insufficient information presented to assess whether or not the ECCS for the 2nd and 3rd generation units is adequate to fully address the demands placed on it for a full spectrum of accident scenarios.

4.1.3. Recommendations

In general our recommendations fall into three categories:

- Implement the proposed modification;

- Perform additional studies or analyses to fully assess the scope of the proposed modification, especially with respect to potential improvements beyond those identified;

- Provide additional information, or analysis, to allow independent verification that all deficiencies have been identified and addressed.

4.2. LOSS OF COOLANT ACCIDENT ANALYSIS

4.2.1. Design modifications

Various design modifications to the ECCS and confinement have been implemented or planned to better protect the plant against a loss of coolant accident. These modifications are discussed in other sections.

4.2.2. Findings

- The RBMK specialists consider the worst design basis LOCA to be guillotine rupture of a pressure header with failure to close the check valve of one distribution group header. The RBMK specialists consider partial pipe/header ruptures to be highly improbable in that any partial break capable of heat removal deterioration is assumed to be greater than the critical crack length and therefore will result in complete guillotine rupture. However, partial breaks could result in periods of flow stagnation in one or more fuel channels which would result in a fuel cladding temperature in excess of those quoted in the reference design basis LOCA analysis. A pipe/header rupture combined with a seismic event could also pose problems if some of the mitigating systems credited are not adequately seismically qualified.

- The use of the 1200°C maximum fuel cladding temperature criterion in assessing the performance of ECCS for a channel type reactor such as the RBMK is conservative and possibly unnecessarily restrictive. The criterion was derived for the typical light water reactor pressure vessel type design.

- Further evidence is needed to demonstrate ECCS equipment and confinement components will not be disabled by cross link effects in the short as well as in the long term. Cross link effects include environmental effects such as flooding, condensation, temperature, pressure and radiation, or dynamic effects such as water hammer, pipe whip and fluid jets.

- The many probability and reliability claims in the reports require reliability analysis support and/or justification of applicability of the data used.

- Further information is needed for the following:
  (a) airborne and possibly waterborne leakage of radioactive materials from areas such as the confinement compartment envelope, safety relief valves and ECCS equipment, particularly long term ECCS equipment;
  (b) assumptions about radioactive releases, such as fission product distribution, washout, plateout and charcoal filter efficiency;
  (c) exposure calculational assumptions such as weather scenario, location and exposure time duration for the most exposed individual.

- For the computer codes used in the LOCA analysis, documentation describing the following is required:
  (a) the physics, mathematical models, empirical correlations and simplifying assumptions;
  (b) validation, including benchmarking against standard or analytical problems, comparison against pertinent experimental data and actual RBMK transients during commissioning and operation and 'blind' testing against relevant test data.

4.2.3. Recommendations

- Channel reactor, PWR and BWR safety specialists should have further discussions with RBMK safety specialists to rationalize the definition of design basis LOCAs for the RBMK and to identify the worst design basis LOCA sequences.

- Channel reactor, PWR and BWR safety specialists should meet with RBMK safety specialists to review whether the use of PWR/BWR ECCS performance criteria in the safety assessment of a channel type reactor such as a RBMK is technically appropriate.

- Separate reviews in the form of specialist meetings should be held to clarify the following subject areas:
  (a) cross link damage;
  (b) reliability analysis;
  (c) radioactive release calculation;
  (d) computer code documentation and validation standards;
  (e) human factor engineering and operator models.

- An independent review of available probabilistic safety assessments (PSA) of RBMK reactors should be performed prior to using the results in support of safety decisions.

4.3. CONFINEMENT

The reactor coolant system of 1st generation RBMKs is not enclosed in a leaktight accident localization system (ALS) similar to the 2nd and 3rd generation plants. Even in the 2nd and 3rd generation nuclear power plants, only a part of the reactor coolant circuit is confined by a system of pressure compartments of an accident localization system. The rooms where the steam/water lines, the steam drum separators and the upper parts of the downcomers are located are not included in the ALS.

4.3.1. Design modifications

For RBMK plants of the 1st generation a decision was already made to construct a separate building housing a pressure suppression system. This building shall be connected to the reactor building. It was said that the pressure suppression system capability for LOCA will be similar to that of the accident localization systems of the 2nd and 3rd generation plants.

Furthermore, investigations regarding the strengthening of the reactor building structure, including the roof of the central reactor hall, are being discussed by the RBMK specialists. Also being discussed is the installation of a leaktight compartment system for the rooms of the steam/water pipelines, the steam separators and the central reactor hall.

4.3.2. Finding

- The defence in depth concept with respect to confining the reactor coolant circuit was not a basis for the design of the 1st generation of RBMKs. The defence in depth concept was also not consequently applied to the 2nd and 3rd generation plants where the upper rooms confining parts of the reactor coolant circuit form practically no leaktight barrier to the environment, especially in the event of a loss of coolant accident in these unsealed rooms.

4.3.3. Recommendations

- The decision to create a pressure suppression system for the RBMKs of the 1st generation with an efficiency similar to that of the units of the 2nd and 3rd generation is supported by the experts. The construction of a separate building seems to be an appropriate way.

- It is recommended to upgrade the upper rooms (reactor hall, steam separator rooms, etc.) with respect to leaktightness and confinement function. It is suggested that the discharge from these rooms in the event of a break in the reactor coolant system could be directed into the pressure suppression system or to a filtered ventilation.

- In accordance with the international practice, consideration should also be given to the installation of steam line isolation valves.

4.4. REACTOR CAVITY OVERPRESSURE PROTECTION SYSTEM

Protection of the reactor cavity against overpressurization is an important safety feature of the RBMK. Cavity pressure exceeding 3.1 atm absolute has been described as having the possibility to lift the upper head/biological shield assembly, breaking the reactor seal, breaking (simultaneously) the pressure tubes, and affecting the operation of other safety features. The existing overpressure protection system has capacity for two or three channel tube ruptures (for first and second generation units, respectively) which reflects a safety margin over the design basis accident of one channel tube rupture. The existing steam discharge system vents the steam/gas mixture from the cavity to a condenser with subsequent gas holdup and release through the gas clean up system/stack for first generation units. For second and third generation units, the discharged steam/gas mixture is discharged into bubbler/condenser pools where the steam is condensed, and gas is retained in the leaktight spaces.

4.4.1. Design modifications

There is an intention to improve the capacity of the cavity overpressure protection system. This work is being conducted in stages which are either completed, under way, or planned for all units concerned.

1st construction stage - A modest increase in capacity amounting to one or two tubes of additional relief is achieved in the first stage by adding two vent valves to existing cavity steam discharge pipes at the top of the reactor. These valves vent the reactor space to atmosphere upon reaching 2.8 atm absolute pressure in the cavity.

2nd construction stage - A further increase in capacity to simultaneous rupture of nine tubes is achieved by addition of a new 600 mm steam discharge pipe into the cavity. This pipe also has two atmospheric vent valves, and additionally is routed to bubbler/condenser pools where they exist.

3rd construction stage - New buildings containing bubbler/condenser pools in sealed areas are planned for first generation units. When existing, the 600 mm steam discharge line would be routed to these pools, eliminating the need for atmospheric vent valves.

4.4.2. Findings

- It is apparent that the modifications being implemented increase significantly the capacity of the overpressure protection system (up to nine simultaneous tube failures). However, this is still small compared to the number of channel tubes passing through the cavity (> 1600), and is even small compared to the minimum number of channel tubes supplied from a common header (43). So the adequacy of the upgrade must be judged in relation to accident sequences having the possibility of resulting in multiple tube ruptures.

- Results of accident sequences analyses were presented by the RBMK specialists. For design basis accident (DBA) sequences, the results indicated that no channel tube failures were to be

expected. The results of a limited set of beyond design basis accident (BDBA) sequences were also presented. Here the initiating event together with an additional safety system failure was analysed and probabilities assigned. The probability of arriving at a core damage state involving multiple tube ruptures was given as 6 x 10^{-7} summed over all the events analysed. The technical basis for this result could not be verified. The conclusion of the RBMK specialists was that there was no design basis for multiple tube ruptures, and the basis for the modifications is to achieve the largest reasonably achievable margin compared to the design basis of one channel tube rupture. A question also remains as to whether there might be other sequences which would be more threatening to pressure tube integrity, especially for multiple tube failures.

- A small net gain in discharge capability is achieved at the expense of an uncontrolled release directly to the atmosphere.

- The reliability of operation of the atmospheric vent valves cannot be established in situ. The range of opening pressures was not described, there was uncertainty over the reliable closing of the valve, and the valve would not be expected to reseal.

- The approach that was presented at the meeting to vent the discharge lines to bubbler/condenser pools precludes the need for atmospheric venting upon completion of the third construction stage.

4.4.3. Recommendations

- The RBMK specialists should proceed with high priority to complete implementation of the improvements to the cavity overpressure protection system.

- Additional analyses should be performed for sequences, based on international experience, that are most severe in terms of pressure tube integrity. The results of these analyses should be subjected to peer review, and selected audit calculations should be performed to support the analysis methodology and results. The purpose is to establish a definitive design basis for the cavity overpressure protection system.

- The boundary conditions for survival of the tubes should be identified, based on a combination of temperature, internal pressure and rates of temperature increase or decrease during the sequences, taking account of any particular features of the tubes such as welds or metallurgical deterioration.

- The RBMK designers should proceed with high priority to provide bubbler/condenser pools at first generation units, together with completion of the second stage of improvement, which will eliminate the need for the atmospheric vents.

- The designer should present results of a series of functional tests on prototypes of the atmospheric vent valves carried out to establish performance characteristics, recognizing their existence only as an interim feature.

5. ELECTRIC POWER SUPPLY

It is understood that comments by the IAEA consultants on RBMK NPP documentation available and meetings with RBMK NPP design and operation engineers were not intended to be an exhaustive safety audit.

The intent of the review was to highlight the existing or potential safety shortcomings in the design of the electrical systems and to suggest ways to improve the general or specific safety situations. The limited information and time available made it possible to cover to a certain depth only some fields of prime safety interest. From the information in the supplied documentation, from the Figs 20-27 in Appendix IV and the discussions, the 3rd generation design of electric power circuits, at all levels from switchyard down to basic battery supplies, appears to be able to meet the usual high level requirements. However, relevant safety aspects were not addressed, such as:

(a) Safety and safety related load coverage by electrical safety and safety related power supply systems;
(b) Sizing and design margins of safety and safety related equipment such as transformers, buses, etc.;
(c) Related situations that could affect safety or safety related equipment, systems or trains separation, brought about by fire exposure, flood, pipe whip or jet impingement effects, seismic interaction of non-qualified on qualified structure and equipment, etc.;
(d) Human factors considerations, such as minimum postulated time before operator's action following accident, and co-ordination actions between different site control rooms;
(e) Testing and maintenance conditions of electrical equipment in general.

There are indications (the Kursk ASSET report for one) that the realization of the design and its long term operation and maintenance will require further study to verify that safety design requirements have been fulfilled.

The changes which were made from the 1st generation design and the backfitting proposals described and illustrated in the next sections indicate an awareness of problems. Some areas of concern which were discussed in varying levels of detail are pointed out.

5.1. DESIGN MODIFICATIONS

Towards the end of the meeting the experts received a list of actions consisting of 14 major items (see Table XVII). Several items had been addressed and are included in our technical notes. Items not covered by the technical notes will be covered by the following comments.

Items 4, 5, 10 and 14 fall into the category of operational reliability with no direct impact on safety systems; they are not commented upon.

Item 6: - "Substitution of leaktight storage batteries for open ones". Not only will the new batteries be leaktight, they will have better seismic characteristics and reliability.

Item 7: - "Substitution of water for gas for fire extinguishing in cable rooms". This action is supported by the experts owing to the better fire extinguishing performance of water based fire fighting systems.

Item 8: - "Cable coating with fire retardant compounds". Cautions on the thermal effect of coating were discussed.

Item 12: - "Improvements in reliability of control circuits for high voltage breakers". After reading the ASSET report for the Chernobyl nuclear power plant to review the root causes of a safety significant accident that occurred on 11 October 1991 at Unit 2, experts were concerned about the following two aspects:

- The high number of cable problems reported;
- Generator No. 3 was reconnected to the power line 30 minutes after the turbine inlet valves were closed due to the failure of the main breaker controls. There is no automatic feature to open the disconnectors when the generator is tripped.

5.2. FINDINGS

- Based on the review performed it was found that some significant differences between the three generations of RBMKs exist. From the first to the third generation of these reactors a systematic improvement in the basic design was noted, e.g., in the areas of safety system redundancy, usage of advanced technical systems and components and additional design requirements against internal and external events.

- The 3rd generation is an evolution over years of experience and has become a satisfactory target design for electrical systems of a nuclear power plant. RBMK designers recognize that the earlier generations must be brought up to a comparable target of safety and reliability wherever possible and the reviewers agree. The urgency with which this is being done was not clear as no schedules of modification plans could be obtained.

- Owing to the complexity, the number of different RBMK generations, and given time restrictions, the reviewers were not in a position to make an overall assessment of the design and layout principles of the electrical system.

- Qualification of electrical equipment under harsh environmental conditions was discussed. It was stated that the necessary equipment was qualified. But further and detailed discussions will be needed to assess this area of environmental qualification of equipment.

- The first generation RBMKs are not equipped with generator circuit breakers, but as part of a backfitting program, such breakers have meanwhile been implemented at Leningrad 1 and 2. Within the subsequent generations of RBMKs, however, these generator circuit breakers are already installed.

- The experts discussed the event of the recent fire in Chernobyl 2 and were informed that the current design is being modified to remotely control the disconnector in the high voltage switchyard from the unit control room. The operating procedure should also require immediate confirmation of the disconnect opening.

- All RBMKs are connected to the main grid via main transformers and standby transformers. Examination of the circuit diagram for the Kursk plant revealed a specific situation in which a single failure of one high voltage circuit breaker can lead to the loss of one turbine generator and its main grid connection at Unit 2 and one standby transformer of Units 3 and 4 at the same time (this, however, does not fail a safety function).

- As a consequence of the Chernobyl 4 accident, the automatic supply of the in-house loads from the main generators isolated from the main grid upon loss of voltage from the grid is now prevented by protective features, where this had been an administrative requirement before, thus causing the diesel generators to maintain power supply to essential loads over an extended period of time until power supply from the grid can be restored.

- Under degraded frequency conditions however, the operator manually reduces reactor power, disconnects both generators from the grid and maintains in-house load by continuing to operate one of the two generators. Transfer from the unit transformer to a standby transformer is always a "slow" transfer ($\approx$ 0.5 s) and special relaying is used to allow critical pumps to continue operation during the transfer.

- An automatic diesel generator startup signal will be initiated on vital bus voltage level of 25% as well as in the case of reactor emergency shutdown.

- Within the second and third RBMK generations the diesel generators serving the safety and safety related systems meet the single failure criterion.

- Information was provided that strict train segregation within the cabling system could not be maintained throughout the plant. Where separation could not be maintained, special features have been used, especially for protection against fire. Further clarification of this aspect will only be possible during site specific inspection.

- A backfitting program is planned for the first generation of RBMKs. A new building, containing three new diesel generators and their associated support systems to supply the safety buses will be built. The existing three diesel generators are intended to supply normal and safety related systems only.

- In the first generation of RBMK the DC supply is from a single battery and one single sectionalized bus, feeding three safety channels. One single failure can disable all safety circuits; however, DC supply could be recovered by manual connection to the other units.

- During the discussions of the DC system the experts were informed about the designed battery discharge time, which presently is 30 minutes, for all the three generations of RBMKs.

- Battery cell selector switches are used within the 1st generation to maintain DC voltage within a specified range.

5.3. RECOMMENDATIONS

- A change for the first generation of RBMKs should be planned with a high priority to increase the degree of redundancy of DC buses and batteries. The degree of redundancy of the DC system should be in accordance with the existing 3 channels of the 6 kV safety supply system. These could be electrically supplied from the three existing alternating current buses. This change could probably be co-ordinated with the large backfit plan for new diesel generators and power circuits, when it is put into effect.

- When relocating the above mentioned batteries into the new "emergency power supply building", no battery cell selector switches shall be used for reliability reasons. Train separation shall be maintained from the new equipment in this building down to the safety equipment inside the existing plant.

- The designed battery discharge time for all RBMK generations currently is 30 minutes, as in other countries in the past. But, meanwhile, it became a standard approach to raise the battery discharge time to the order of 1 hour and more to cope with station blackout (simultaneous loss of all off-site and diesel generator AC power sources) and accident management requirements. The experts therefore recommend the implementation of batteries of higher capacities of the order of 1 to 2 hours, based on an analysis of system behaviour under station blackout conditions. The priority of this work is dependent on the priority assigned by the Accident Mitigation Group.

- The undervoltage criterion for automatic diesel generator startup should be reanalysed (80% of the nominal voltage is usually used in western plants). An optimization investigation should be carried out to avoid frequent diesel startup caused by transient voltage fluctuations of the grid. This recommendation falls under the second level of priority.

- Redundant channel separation and protection against common mode failures should be further analysed on a site by site basis; this analysis should also be given high priority.

- Further investigation concerning qualification of electrical equipment under harsh environmental conditions should be performed.

Appendix I
CORE MONITORING AND CONTROL

I.1. RADIAL AND AXIAL POWER DENSITY DISTRIBUTION

Scope of review: Power distribution;
Allowable peaks;
Power sharing.

Document/section reviewed: [1]

I.1.1. Summary of discussions

The uncertainties in calculating power distribution in critical experiments on fresh fuel are 5 - 10%; at full power the uncertainties range from 15% up to 30% (peripheral fuel assemblies (FAs)).

TABLE III. POWER DISTRIBUTION AFTER LOADING 2.4% ENRICHED FUEL

Reactor Type	f_R	f_{AX}	Comments
RBMK 1000	1.4 - 1.45	1.15 - 1.2	double hump skewed towards bottom
Smolensk 3	1.25 - 1.3	1.3 - 1.35	average axial peak 1 m below mid-plane of core
RBMK 1500 (Ignalina)	1.3 - 1.4	1.3 - 1.4	double hump peak above mid-span high peak below mid-span

Note: f_R; f_{AX}: radial power and axial peak power over average power.

These values seem to be actual values.

TABLE IV. MAXIMUM CHANNEL POWER AND LINEAR HEAT RATE FOR DIFFERENT PLANTS

Reactor Type	Max. Channel Power	Max. Linear Heat Rate
RBMK 1000	2.9 MW	300 W/cm all units except Kursk 2,3
RBMK 1500 (Ignalina)	3.6 MW	400 W/cm

The calculation concerning the refuelling in the core is done at the plant. The plants are responsible for establishing their refuelling plans and the refuelling procedures.

The calculation is performed using the codes STEPAN in Ignalina, and by the codes OPERA for RBMK 1000.

Another set of calculations is being performed to determine power distributions. These calculations are done by the 2-D code BOKR. The results of this calculation are fed into the PRISMA code, which is used for the on-line power reconstruction done on the SKALA system.

Before the Chernobyl accident the calculation used a database generated for each fuel assembly by the WRM code. Now the database is generated using the WIMS code.

The results of the calculation performed by the SKALA system of the power distribution are continuously compared with the power distribution generated on the basis of the in-core detectors.

The power distribution for the transient analysis is calculated by the RDIPE which also performs safety and transient analysis. The transient analysis is performed on a worst case basis (conservative initial and boundary conditions).

Power distribution, reactivity worth of individual control rods and banks, and reactivity coefficients determined in experiments are the basis for the comparison with the transient analysis. The burnup is taken into account by a special procedure.

Maximum allowable peak

For RBMK 1000, the operational limit for the channel power is 2.9 MW. The maximum allowable channel power is 3 MW which is also the value used in the safety analysis. Maximum axial peak power and channel power are correlated by the formula:

$$f_{AX} \leq 4.2/W$$

where W is channel power in MW (see Tables III and IV). The maximum axial peak power must not exceed the value of 1.7.

If the operational limit for the channel power is exceeded by 5-10%, the operator gets an alarm and is supposed to manually reduce power in the core zone concerned.

If there are two or more channels outside the allowable limits, the reactor power will be decreased manually until the signals disappear.

Conversion of in-core signals to power

The individual signals are summed up and normalized. PRISMA results are presented on a display and compared with the direct readings.

For calibration purposes of the in-core detectors, the heat balance of the total plant is used. This calibration is performed once a week. Detector burn up is taken into account by special procedures.

Power distribution

During the transition period in going from 2.0% enriched fuel to 2.4% enrichment, a power distribution problem had to be overcome.

Now, with the majority of the fuel assemblies of the higher enrichment, the radial peaking problems have been normalized.

I.1.2. Finding

1. In case the allowable limit of channel power is exceeded, the operator is supposed to reduce power in the channel. If he fails to do so, the local control and protection systems will act.

I.1.3. Plant specific status

Not applicable.

I.1.4. Recommendations

1. Efforts to install 3-D methods for the analysis of power distribution as a function of burnup should be continued

2. A 3-D off-line computer program to predict power and burnup distribution should be developed.

3. Methods should be independently assessed in more detail.

I.2. SUBCRITICALITY OF SHUTDOWN AND UNPOISONED REACTORS

Document/section reviewed: [1]

I.2.1. Summary of discussions

Reactor power control and shutdown is achieved by 211 control rods, 24 of which are part of the fast acting scram system (FSS). They provide a total reactivity worth of 2.5β. Control rod worth is given in Table V.

TABLE V. ROD WORTH

Reactor Type	Reactivity worth hot shutdown		Reactivity worth cold shutdown, unpoisoned (requirement)
	211 Rods	24 Rods (FSS)	
1st Generation	10 - 12β	2β	4β (2 %)
2nd Generation	12 - 15β	2.5β	4.5β

Each rod on the average has a reactivity worth $\Delta k = 0.1\beta$.

The regulatory requirement for cold shutdown in case of repair with FSS rods withdrawn is not less than 4β (2%) instead of formerly 2β (1%).

The "stuck rod" is considered in the calculation.

The enrichment (2.4% U-235) has a negligible effect on the rod worth according to the Russian calculations.

With partial installation of new control rods into RBMK 1st generation reactors, the cold shutdown margin is not less than 3β. First requirement: maintain peak to average neutron flux ratio <3.5 under cold conditions. Second requirement: maintain flux ratio <2.5 under hot conditions with rods withdrawn.

I.3. CONTROL AND PROTECTION SYSTEM AND THE OPERATING REACTIVITY MARGIN

Document/section reviewed: [1]

I.3.1. Summary of discussions

Most of the discussion about control rods was devoted to the methods used for the on-site calculations of operative reactivity margin (ORM). Some attention was paid to some exceptional reduction of ORM during transition to lower power.

The relationship between control rod insertion and void coefficient was also considered. Finally, the methods used for evaluating the cross-sections in the control rods assemblies were discussed. The type and number of control rods in RBMK reactors are given in Table VI.

The RBMK designers have implemented certain changes to the CPS and ORM for RBMKs. Other changes have not yet been implemented.

TABLE VI. TYPE AND NUMBER OF CONTROL RODS IN RBMK REACTORS

Unit	CR	MCR (new-design MCR)	SBR	AR	LAR	FSS
Chernobyl 1	179	118 (21)	21	12	7	21
Chernobyl 2	179	118 (21)	21	12	7	21
Chernobyl 3	211	131	32	12	12	24
Kursk 1	179	117 (21)	21	12	8	21
Kursk 2	179	117 (21)	21	12	8	21
Kursk 3	211	135	32	12	8	24
Kursk 4	211	135	32	12	8	24
Kursk 5	223	150	32	8	12	21
Leningrad 1	191	127 (21)	21	12	7	21
Leningrad 2	179 (191)	118	21 (32)	12 (8)	7 (12)	21
Leningrad 3	211	131	32	12	12	24
Leningrad 4	211	131	32	12	12	24
Smolensk 1	211	131	32	12	12	24
Smolensk 2	211	131	32	12	12	24
Smolensk 3	211	134	32	12	9	24
Ignalina 1	211	131	40	4	12	24
Ignalina 2	211	131	40	4	12	24

CR = Control rods
MCR = Manual control rods
SBR = Short control rods inserted from the bottom of the reactor
AR = Automatic control rods for controlling the power level
LAR = Automatic control rods for controlling local power
FSS = Fast scram system
( ) = after modification

The changes which have been made already (summarized in Table VII) are as follows:

- Absorber assemblies (AA) are now left permanently in the core. About 80 AAs for RBMK-1000 and about 50 for RBMK-1500 are now in place.

- Modification to present control rod design to prevent the possibility of positive reactivity addition when rods move downwards.

- 8 control rods were changed for 2nd generation RBMKs so that they now enter from the bottom of the core. This brings the total of bottom rods from 24 to 32.

- All bottom rods are now inserted when a scram occurs.

- The number of effective manual control rods (MCR) for the ORM is now about 45 for RBMK-1000 and about 55 for RBMK-1500.

- The ORM is now calculated every 5 minutes instead of every 15 minutes.

- The rod insertion time for the control rods is about 12 seconds instead of 19 seconds (more than $2\beta_{eff}$ in less than 3 seconds from 24 fast acting rods).

TABLE VII. SUMMARY OF RBMK DESIGN CHANGES TO THE CONTROL AND PROTECTION SYSTEM

Unit	Number of Absorber Assemblies (AA)		Modified Control Rod (CR) Design		Number of CR of New Design		Number of Bottom Rods (BR)		BR Move on Scram	
	before	after	before	after	before	after	before	after	before	after
Chernobyl 1	0	84	No	Yes		21	21	21	No	Yes
Chernobyl 2	0	108	No	Yes		21	21	21	No	Yes
Chernobyl 3	0	93	No	Yes			24	32	No	Yes
Kursk 1	0	101	No	Yes		21	21	21	No	Yes
Kursk 2	0	103	No	Yes		21	21	21	No	Yes
Kursk 3	0	87	No	Yes			24	32	No	Yes
Kursk 4	120	81	No	Yes			24	32	No	Yes
Kursk 5	0			Yes				32		Yes
Leningrad 1	0	80	No	Yes		21	21	21	No	Yes
Leningrad 2	0	80	No	Yes		21	21	21	No	Yes
Leningrad 3	0	81	No	Yes			24	32	No	Yes
Leningrad 4	0	80	No	Yes			24	32	No	Yes
Smolensk 1	0	81	No	Yes			24	32	No	Yes
Smolensk 2	0	81	No	Yes			24	32	No	Yes
Smolensk 3	0	99	No	Yes				32	No	Yes
Ignalina 1		52	No	Yes			24	40	No	Yes
Ignalina 2		54	No	Yes			24	40	No	Yes

Note: Before and after refers to Chernobyl accident.

TABLE VII. (cont.)

Unit	Control Rod Insertion Time (approx) (s)		Operating Reactivity Margin (ORM)		ORM Calculation Frequency (min)		ORM Violation		Number of FA, 2.4% Enrichment
	before	after	before[a]	after[b]	before	after	before	after	status 1992
Chernobyl 1	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	891 (Feb)
Chernobyl 2	18	14(+3,+2)	>26	43-48	30	5	Admin.	Admin.	946 (Feb)
Chernobyl 3	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	986 (Feb)
Kursk 1	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	962 (Sept)
Kursk 2	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	979 (Apr)
Kursk 3	18	14(+3,-2)	>26	43-48	30	5	Admin	Admin.	1169 (Sept)
Kursk 4	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1357 (Sept)
Kursk 5		14(+3,-2)					Admin.	Admin.	
Leningrad 1	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1372 (Aug)
Leningrad 2	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1469
Leningrad 3	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1557 (Aug)
Leningrad 4	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1557 (Aug)
Smolensk 1	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1443 (Aug)
Smolensk 2	18	14(+3,-2)	>26	43-48	30	5	Admin.	Admin.	1429 (Aug)
Smolensk 3	18	14(+3,-2)		43-48	30	5	Admin.	Admin.	628 (Aug)
Ignalina 1	18	14(+3,-2)		53-58	30	5	Admin.	Admin.	
Ignalina 2	18	14(+3,-2)		53-58	30	5	Admin.	Admin.	

[a] it is checked with listing.
[b] it is checked with self-recorder, digital device, display.

One change to the CPS which is being implemented slowly is the replacement of control rods with a new design (it is not considered essential by RBMK specialists to do it promptly since water has been removed from the existing rods). Another change, which is an administrative change, is that the operator shall make a manual scram if the ORM is violated.

A typical RBMK design which contains 211 control rods has the rods distributed as shown in Table VIII.

TABLE VIII. DISTRIBUTION OF CONTROL RODS

Type	Number	Function
Manual control	131	Operator controlled - a portion is used to shape power and a portion is reserved.
Local automatic regulation	12	Maintain power shape by using signals from in-core detectors.
Automatic power regulation	12	Maintain total reactor power by using signals from four lateral ionization chambers. Three banks of four rods.
Scram	24	Scram rods - normally withdrawn from core.
Short absorbing	32	Used to control axial power shape - manually controlled, and enter from bottom of reactor.

I.3.2. Findings

1. A systematic classification of the control rods with respect to the role (control, protection, emergency) and to the moving mechanism (manual, automatic, fast acting scram system) in each RBMK-1000 of first and second generation and in the RBMK-1500 was not provided; it should be available at all plants concerned.

2. Information on the position selected for the 21 redesigned control rods in the RBMK reactors of the 1st generation was not sufficient.

3. The redesigned control rods will also be implemented in the second generation RBMK. No information was provided on the status of implementation.

4. Some concern comes from the reactivity effect related to the loss of cooling of control and protection system (4-5β, in future 2-3β): the concern is increased by the fact that only one cooling circuit exists. Nevertheless, protection against loss of water is assessed by 3 parameters, related to pressure in the header, water flow rate and level of water in the upper tank, which initiate water makeup to systems with breaks.

5. All of the fixes which are made to the CPS and ORM appear to be good and should be in the direction of improving the safety of RBMK reactors. Overall, these design changes should increase the safety of the reactors in many respects, but it was not evident to the western experts that the proposed changes will not reduce the safety margins in other respects. The material presented does not include all the details of the analyses used to make this trade-off between increased safety and possible reduced safety margins. It is recognized that the RBMK reactor is very large, and the chain reaction in one part of the core is only very loosely coupled with that in other, distant regions. It is a requirement to control the spatial power distribution almost as if there were several independent reactors within the core volume. This situation can lead to conditions where small spatial redistributions of reactivity can cause large spatial redistributions of the power. The proper assessment of the real trade-off of sometimes competing and interactive design fixes requires three-dimensional analyses.

6. As seen in Table VIII, a complicated combination of manual control and automatic control is utilized to maintain total reactor power, as well as the power shape. The SKALA central monitoring system, as well as other instruments, provide the operator with information concerning the power level and power shape. The SKALA system provides updated information every 5 minutes. The system is designed to provide guidance to the operator for steady state control of the power distribution. Under transient conditions, this system would be much less effective.

7. The SKALA system also calculates the ORM. The ORM is expressed in terms of the number of "equivalent" control rods remaining within the core. This definition is not precise, but the ORM is very important to safety because of its effect on the void and power coefficients.

8. It was also noted that when a violation of the ORM occurs, the operators are only required to scram the reactor manually by following an administrative procedure, since this has not been implemented into an automatic engineered safety feature.

I.3.3. Plant specific status

See Tables VI, VII.

I.3.4. Recommendations

1. The modifications introduced should improve safety, but may lead to possible reductions in safety margins. Possible examples are:

   (a) The increase to 2.4% enriched fuel may have a negative effect on power peaking;

   (b) The addition of fixed absorbers may have a negative effect on the shutdown margin.

   Because of the complicated nature of the RBMK reactor physics, an independent assessment of these sometimes competing and interactive design changes should be conducted using 3-D analyses. Results of 3-D analyses performed by RBMK and western specialists should be compared.

2. Backfitting of additional instrumentation and control devices should be encouraged in all areas, where such additions can be shown to improve safety. Possible examples are:

   (a) In first generation RBMKs, graphite temperature is measured at 24 locations (8 radial by 3 axial). In second and third generation RBMKs, graphite temperature is measured at 74 locations (6 radial by 5 axial, plus 11 radial by 4 axial). Graphite temperature in the remainder of the core is provided by calculation with the SKALA system. A possible improvement to safety would be to provide additional instrumentation to measure more graphite temperature locations; and

   (b) To increase the computing power of the SKALA system to allow more frequent updates to the reactor status than the present five minute interval. Also the information provided to the operators in transient conditions in RBMK and relevant western plants should be compared.

3. Due to the importance of the ORM to safety, its violation should be implemented into an automatic engineered safety function. High priority should be assigned to ongoing work in this area.

4. The replacement of the ORM concept, which requires too much operator involvement, should be further explored. The use of additional and/or improved automatic instrumentation and control equipment is suggested.

I.4. METHODOLOGY AND CODES FOR NEUTRONIC CORE DESIGN

Document/section reviewed: [1], [2]

I.4.1. Summary of discussions

The evaluation scheme used now consists of two main steps. The first is aimed at elaborating cross-sections for core calculations. The second step is devoted to core calculations (power distribution and reactivity).

The two groups cross-sections are performed with WIMS D4, the nuclear data are elaborated with NJOY from ENDF B 4 and 6 files. The fuel assemblies are calculated for an infinite medium for given parameter values such as fuel temperature, water density, graphite temperature, enrichment,

xenon concentration. The absorber characteristics are obtained assuming a cylindrical cell with an external multiplying zone. These cell evaluations are performed in the Kurchatov Institute, in RDIPE and VNIIAES. The validations are made by comparison with Monte Carlo codes MCNP (USA) and MCU (Russia).

The core analysis is carried out in Kurchatov Institute and RDIPE with different tools. The core is modeled with 1 mesh per assembly radially and, currently, 16 meshes in an axial direction. These codes used the finite differences method. The validation is obtained by comparison with critical experiments and plant measurements; for the power density the discrepancies reach 10% and 30% respectively.

On the plants, calculations for the full core are based on 2-D and 2 group diffusion theory; the influence of an inserted rod is evaluated with 3-D calculations around the control rod.

To perform flux reconstruction, the plant calculation scheme used also two groups and 2-D codes.

I.4.2. Plant specific status

Not applicable.

I.4.3. Recommendation

1. Proper core calculations including 3-D tools are recommended to be available directly at the plants. Particular attention must be paid to the neutron field calculations near the reflectors.

I.5. REFUELLING PROCEDURE

Scope of review:	Procedure for channel refuelling; Change of reactivity; Deformation of neutron fields during refuelling; Control rod action.
Document/section reviewed:	[1]

I.5.1. Summary of discussions

On-power refuelling is carried out at the rate of 1.4 fuel assemblies per day on the RBMK-1000 and 1.8 - 2.0 fuel assemblies per day on the RBMK-1500 reactor. Documented procedures for channel selection and fuel loading exist and are approved by the chief engineer or his deputy in charge of nuclear safety.

Power distributions in the core are calculated with a simple 2-D code with a synthesized axial distribution (new 3-D codes for fuelling are now being developed). From this power distribution, channels for refuelling are selected on the basis of highest burnup and other boundary conditions which are documented, but not presented at this meeting.

The highest burnup channel may not be selected if the neighbouring channel contains fresh fuel. The overall strategy is to maintain core symmetry with roughly equal number of control rods inserted in each quadrant. The maximum channel power limit for the refuelled channel is 3.0 MW with the neighboring channels kept below 2.3 MW.

New fuel in a channel causes a local flux/power increase which is compensated initially by automatic control rod insertion followed by operator action to insert manual control rods as far as possible. Automatic control rods move to compensate for xenon buildup following refuelling. The regulators review and approve the procedures used in the refuelling process for each plant.

Once a channel is selected for refuelling, the old assembly is removed, the flow manually readjusted and the new assembly inserted. The flows are adjusted at midlife between fuelling (fuel assemblies remain inside typically for 3 years). The critical heat flux margin seems to be between 1.3 and 1.5.

I.5.2. Finding

1. The procedures/practices as described are comparable to those used for on-power refuelling in Canada.

I.5.3. Plant specific status

Practices as described are applied at all units.

I.5.4. Recommendations

1. Development of 3-D code for calculation of power distribution should continue.

2. Independent review of the documents and procedures referred to by the RBMK specialists should be undertaken to confirm the adequacy of procedures.

I.6. COEFFICIENTS AND EFFECTS OF REACTIVITY

Document/section reviewed: [1], [2]

I.6.1. Summary of discussions

The matter of the discussion was mainly related to methodology for neutronics (see the corresponding topic).

I.6.2. Findings

1. Complete information on the experimental measurement of reactivity coefficients and, in particular, void reactivity coefficients was not available for the review.

2. Relationship between burnup and void coefficient is important. There were no reports presented on this topic at this meeting but RBMK specialists discussed this subject in a paper presented at the IAEA in April 1992. In an RBMK-1000 using 2.4% enriched fuel with at least 80 additional absorbers installed, the discharge burnup is about 20 MW.d/kg U. With these conditions the value of the void reactivity is about 0.8β.

3. Information on consideration of the samarium at start up was not available for review; it will be discussed at future meetings.

I.6.3. Plant specific status

Not available.

I.6.4. Recommendations

1. Measures to allow for calculation of the reactivity coefficients in any real situation directly on plant sites are strongly recommended.

2. The possibility of void reactivity reduction should be further explored in areas of both design and operation (a similar suggestion was made earlier by the Lithuanian regulatory body for the Ignalina plant).

I.7. CORE MONITORING AND SHUTDOWN

Scope of review:	Power monitoring; Coolant flow monitoring; CPS monitoring; Monitoring of other process parameters.
Document/section reviewed:	[1], [2], [3]

I.7.1. Summary of discussions

The major part of the discussions was directed at clarifying the material contained in the two reports and reconciling that information with the reviewers' understanding and existing knowledge of the RBMK reactors. The major difficulties arose from the significant differences that exist between RBMK reactors currently in use. In general it appeared that the monitoring and protection systems were much better organized and more comprehensive in scope than was apparent from the prepared material. In the discussion it was not often possible owing to time constraints to obtain a detailed understanding of the design intent. Two examples of this are given: first, in the case of reactor control it was not clear if the plant was expect to follow or try to force the external grid; consequently the plant control strategy could not be determined. In the second case, the protection system, it was not clear how the action of the shutdown systems, the signal processing and the choice of set points related to the safety studies. The safety requirements as referenced in the regulatory documents were not known to the reviewers.

The in-core instrumentation system to monitor power density distribution was discussed along with ex-core systems.

The core coolant flow monitoring system was discussed but no details of the discussion are recorded here.

Instrumentation and the system to measure and display control and scram rod position was discussed; no details of the discussion are recorded here.

Instrumentation to measure general process and thermo-hydraulic core/primary circuit parameters (temperature, pressure, flow rate) was covered, but not in depth. No details of the discussions are recorded here.

Reactor power monitoring (neutron flux) was also covered.

Control of reactor power and its distribution were discussed in general but power limitation depending upon equipment in operation was not covered in detail owing to time constraints.

Transfer of the reactor core to subcritical conditions and maintenance of subcriticality was not covered.

The scope of the review did not include discussion of the architecture of the control and protection systems, the choice of technology or the justification of the reliability of the systems. This restriction of the scope was necessary because of the complexity of the systems and the limited time available.

I.7.2. Findings

Control strategy

1. The overall strategy adopted for the control of the plant and consequently of the core was not clearly stated at the meeting. The consequences of the strategy were that core power output was set and maintained and the main coolant flow to the core was held constant. The measuring system measured the flux, not the power; however, in all discussions about the control systems the two were taken to be identical. All plants use ex-core detectors to control bulk power during startup. The local systems are brought into use once the reactor is at or above 20% of nominal design power.

2. The ex-core detector based bulk power system makes use of ionization chambers at four stations around the core to control the axial displacement of a bank of four rods. It was established that there may be up to three banks of four rods on a plant, one in use, one in reserve and one for use at very low powers, i.e. less than 5% of nominal power. Each bank of control rods is associated with a set of a few ex-core ionization chambers. The arrangement of the ex-core detectors was found to be plant dependent. The reactors have 24 channels for ex-core instruments, the detectors operating the 3 banks of rods occupy 12 of the stations, the remaining 12 stations being taken up by detectors associated with shut down and measurement. The axial arrangement of the ex-core detectors was a single chamber at the core mid-plane on RBMK 1000. The channels at Ignalina have chambers at 3 stations, all reactors are moving over to this arrangement. It was reported that the ex-core system was originally used for core power control on all RBMKs. One of the early ex-core systems is in fact still used on the four Kursk reactors. The Kursk arrangement for local power control is that the core is divided into 8 segments. The power in each segment is controlled by a single control rod and a single ex-core detector. This arrangement would appear to be less sensitive to changes than the local control systems using in-core detectors. A second form of control system based on ex-core detectors was said to be the secondary control system in the Ignalina plant. In this case the control system would appear to comprise two banks of four control rods, each bank being controlled by a group of four ex-core detectors.

3. Three arrangements of azimuthal and radial power control system using in-core detectors were identified as being: seven zone, nine zone and twelve zone control systems. All three systems use the output of in-core self-powered detectors to control the adjustment of position of a single control rod in each zone. The number of detectors in each zone varied by plant and could be two, four or six. The treatment of failed detectors by the control system could not be clarified in detail. Equipment condition monitoring was said to be done but the form of the monitoring and subsequent action could not be established.

4. The control algorithm was found to be that simple rod motion was started when the power deviation was larger than 1.0 % and stopped when the deviation fell below 0.95%. There are

no limits on interzone power differences. However, a range of actions was reported as being possible in the event of a large power deviation. For 7% overpower difference between the zone power and the set-point, the regulation of power stops. If the difference rises to 10% then a power setback was invoked, SS-6, which stops when power deviation goes. The operator was said to be responsible for taking action in the event that the power in a zone fell out of range. The impact of the change from delayed silver emitter to prompt hafnium emitter in core detectors on the control algorithm was not discussed.

5. The control of the axial power distribution is by operator action using the bottom and top entry manual control rods in conjunction with information from the SKALA system and the in-core detectors.

6. The move to a common 12 zone control system using in-core detector is agreed to be a good action; however, the implications of the change on plant currently using other systems as reviewed by the RBMK specialists should be examined.

Monitoring

7. The monitoring of plant parameters and processing of plant information was found to be extensive. The SKALA and TITAN computer systems would appear to be very heavily loaded and currently have a slow cycle time, e.g. update of flow every 40 seconds and dryout margin every 5 to 15 minutes. The efforts and actions to accelerate the rate of data processing and information updating are supported by the experts.

Shutdown

8. The main changes since Chernobyl related to the protection functions based upon neutron flux measurement. This appears to be because the Chernobyl accident was a neutronics accident which appears to have guided much of the backfitting work. The change of manual shutdown button from one that had to be manually held down until the plant was shut down to one that need only be pressed once is most important. The action to reduce the times for rod insertion and the speed with which the reactor is shut down is supported. The tables giving the shutdown algorithms in supporting document Shutdown Systems, Ref. [3], require further study by the reviewers and the link of the triggering action to the accident transients and to the plant safety case must still be made. The approach to proof and periodic testing appeared similar to those employed in western plants and the role of such testing when making the very strong claims for shutdown system reliability should be further discussed. The development of the ASS, additional scram system, with an effective of inserting 2β to 3β in fractions of a second was noted. The effect of such a rapid scram on the integrity of the structure arising from thermal and hydromechanical shock is worthy of investigation.

General

9. The division of the control rods of the same function among the six power supplies was seen as good practice but there remains the question to what extent this segregation is carried through to cable runs and the location of processing equipment. For example it was identified that the power and doubling time shutdown instruments were located in the same room on Smolensk and Ignalina. This practice differs from that adopted in some other countries. There are plans to improve segregation in future systems.

I.7.3. Plant specific status

To be expanded, in particular, to identify what detectors, controllers, back up controllers, set-points, etc., are in use on what plants.

I.7.4. Recommendations

1. More information on the control strategy adopted for RBMK reactors is required.

2. More information on the protection strategy adopted for RBMK reactors is required.

3. It is proposed that further work be undertaken to establish the requirements of the plant safety case and safety arguments and the links to the protection functions that are and will become available on the plant.

4. In the long term it would be desirable to link the protection function to what it is intended to protect and what the protection system has to achieve, e.g. in terms of absorption insertion, to provide protection.

5. The tables of the supporting document Shutdown Systems, Ref.[3], should be expanded and it is suggested that the tables are presented in two ways. First, each shutdown mode FSS, SS, etc., should be identified, then each case, i.e. set of parameters, which cause that system to be invoked should be listed for each of the RBMK reactors. Second, the data could be presented as in Table 5 of Ref.[3]. Tables should be expanded to indicate the differences on the different reactors and to include all other parameters that cause plant shut down.

6. It is proposed that a core plan be produced for all reactors showing the location of the fuel, different detectors and different control rods.

7. It is suggested that the implications of the change in azimuthal power control with regard to change in detector type and introduction of 12 zone controllers be reviewed.

8. The impact of the use of the very fast acting ASS on the system integrity needs review to ensure that it introduces no unwanted effects.

9. The data processing computers should be upgraded and additional warning introduced.

I.8. REACTIVITY ACCIDENTS

Scope of review:	Core behaviour transients for 3 classes of reactivity initiated accidents in RBMK reactors.
Document/section reviewed:	[4]

I.8.1. Summary of discussions

RBMK specialists presented a brief description of the methods and assumptions used for calculating the neutronic dynamic behaviour of the RBMK core for accidents causing reactivity addition. These fall into three categories:

- Changes in reactivity due to significant and abrupt changes in the core coolant density (ultimate design base accident-UDBA, false actuation of ECCS, gas ingress from ECCS into a fuel channel, etc.);

- Changes in reactivity due to the unauthorized withdrawal of CPS rods or under refuelling;
- Changes in reactivity due to water loss in the CPS circuit.

Accordingly, the discussion focused mainly on three accidents:

- The UDBA, which is defined as MCP pressure header rupture in one of the reactor halves;
- Control rod withdrawal (since these accidents add reactivity quickly);
- Loss of water from the CPS.

In order to model these accidents it is necessary to use 3-D models for the neutronic calculations taking into account thermal-hydraulic feedback effects. Since their code validation is not complete they calculate the core neutronic behaviour using several codes - TRIADA, STEPAN, TREP and RADA - as a means of estimating potential error. Furthermore they have performed a variety of measurements at several power reactors and have concluded on the basis of these measurements that the codes are sufficiently accurate for all conditions considered.

The neutronic and thermal hydraulic calculations are necessarily iterative. The neutronic distributions for the high power regions of the core are calculated using codes such as RELAP-4 and TRANS-8 to determine the voiding rates of the broken portion of the core. Boundary conditions applied to the models are conservatively chosen for the accident scenario being studied. The UDBA scenario is considered from a variety of initial power levels and from a variety of assumed ORM conditions.

The control rod withdrawal accidents use models employing quarter core symmetry and look at withdrawal of every rod. The characteristics of the CPS systems for 7, 9 and 12 zone protective systems are considered.

Reactivity additions from CPS cooling circuit failures require detailed calculations since the voiding rate depends on the number and position of rods in the core. While the reactivity addition is considerably slower than the previous cases considered, the maximum void effect is about 4ß, which exceeds the capability of the FSS.

I.8.2. Findings

General

1. The RBMK reactor, by its characteristics and features, falls into the category of large core reactors in which spatially distributed processes are in many cases important to the analysis of safety. The dynamics of the reactor are such that local power distortions can result in severe consequences unless special monitoring, control and safety systems are provided. It is important to note that there have been significant improvements to these systems on RBMK reactors since the Chernobyl accident and further improvements will result from full implementation of the "Modernization Plan".

2. Since the Chernobyl accident there have been several modifications which have resulted in a reduction of the positive void coefficient and mitigation of its effect on accidents:

   - introduction of approximately 80 additional fixed absorbers;
   - increase in fuel enrichment from 2.0% to 2.4% (currently being introduced on all plants except Ignalina);
   - increased ORM and increased minimum ORM;

- introduction of 21 to 24 fast acting control rods which constitute the FSS with an insertion time of 2.5 s;
- increase in the number of bottom inserted rods to 32 rods;
- introduction of new control rod design to eliminate the "positive scram" effect which contributed to the Chernobyl accident;
- increase in speed of insertion of control rods from about 19 s to 12 s.

3. The reduced positive void coefficient, in addition to having a direct effect on the consequences of the accident scenarios considered below, has resulted in the reactors now having a negative power coefficient over their entire power range.

4. It should be noted that references to full void effect mean void from full power operating conditions rather than from full to empty channel conditions.

Ultimate design basis accident

5. Using conservative initial conditions such as minimum ORM of 30 rods, fuel enrichment of 2.0%, fastest channel voiding time, etc., the analysis shows that the resulting void for both loops voiding from zero power hot conditions is 2.2ß. For one loop voiding, the reactivity added under emergency shutdown system failure condition is 0.8ß (when feedback effects are taken into account) starting from ORM equal to 30 rods at nominal power conditions. The corresponding value for both halves of the reactor blowing down has been calculated but was not presented at this meeting. These values are lower than the negative reactivity inserted by the FSS rods(2.5ß)

6. The results given in Section 2.2. of this report give a void reactivity coefficient of 1.7ß. With an ORM of 45 rods, the coefficient is 1.2ß. The actual coefficient has been measured as 0.8ß with the 2.4% enriched fuel which has been added to date.

7. The local power pulses resulting from half-core voiding considering only FSS rods with the most effective one missing reach values of about 3 times nominal power in the region of the FSS rod failure.

8. These calculations take into account electronic delays (0.1 s) and actual fast rod drop times (2 to 2.5 s) as measured both experimentally and at NPPs. The electronic delay time seems small.

9. The "standard CPS is capable of providing reliable suppression of neutronic power", limiting integral reactor power to 125% of rated power and short term increase in power density in the lower part of the core to 60% of the initial value.

10. All reactor power levels have been considered in the calculations (even below 20% where power operation is forbidden) and the results show power pulses no higher than quoted above.

11. The calculation for fuel temperature based on these calculations is less than 1200°C and only a very small amount of fission products is released. This result is also dependent on the effectiveness of the ECCS, which was not discussed here.

12. On the basis of the results presented, there has been a significant improvement in safety system performance since the time of the Chernobyl accident. It appears that in general the type of calculations presented is similar to those used in the analyses of the CANDU reactor. Detailed review of the documents listed as references (but not distributed at the meeting) will be necessary to confirm this. The power peaks quoted above are also comparable.

Rod withdrawal accidents

13. This set of accidents, introduced into safety analysis of the RBMK reactor after the Chernobyl accident, considers spontaneous withdrawal of individual control rods. Since these are dispersed throughout the core, the need for 3-D neutronics calculations in the consequence analysis is obvious. Once the neutronics distribution is calculated, RELAP 4 (now changing to RELAP 5-MOD 3) is used to calculate the thermal-hydraulic conditions in the channel of interest.

14. In order to provide a conservative estimate of the possible consequences of these accidents, very conservative starting assumptions are used. It is assumed that two neighbouring channels have been refuelled (a violation of fuelling rules), the rods are fully inserted, once the rod starts it continues withdrawing to end of travel (mechanical failure of stopping devices) and the operator does nothing to stop the rod.

15. Rod withdrawal causes a peak in the neutron flux in the vicinity of the withdrawn rod. If this type of accident is analysed on the assumption that the control and protection system works, then the flux distribution changes due to insertion of other rods in the vicinity. Depending on the specific conditions, the flux distribution can actuate either the LSS or the power scram system (PSS). The effectiveness of the response of these systems depends upon whether there is a 7, 9 or 12 zone control system and where the location of the rod is relative to the detectors. In the case of Kursk, there are only ex-core detectors, so coverage will be poorest in that reactor.

16. The results presented show that for the majority of cases considered, the aims of maintaining the cladding temperature below 1200°C and preventing fuel melting are achieved. Only in the case of all the above mentioned conservatisms acting together for those reactors with 8 zone control systems does very limited fuel melting occur in one channel. Therefore for these reactors an administrative procedure has been instituted which requires that the drive mechanisms be electrically disconnected when some of the control rods are fully inserted. This procedure is followed at Kursk NPP. In order to check the accuracy of the calculational methods used, an experiment was conducted, at 50% full power, whereby a rod was withdrawn and the nearest zone automatic regulating system was switched off. The results showed an increase in power by a factor of 1.6 and the maximum error in the calculated peak was approximately 6%.

17. The modernization measures planned, which will activate the FSS system from in-core detectors, thereby allowing better coverage for rods away from the periphery of the core, will further improve the safety of the RBMK reactors for this type of accident.

CPS cooling circuit failure

18. Loss of cooling water in the CPS channels will have a distributed and non-uniform effect on the core. The specific rate of water loss depends upon the CPS rod design and their position in the core since the rate of leakage is governed by how much water is in the channel. The rod design and the area of the telescopic tie connecting the absorber and displacer are such that simultaneous loss of water in all channels where the telescopes are used leads to a positive insertion of reactivity of 4ß. However, the results of the analysis presented show that maximum rate of reactivity addition is 0.125ß/s, which can be handled by the scram system.

19. Acceptable results from the analysis of this event are clearly dependent on the accuracy of the codes used to predict the rate of discharge of water from the channels. This needs to be confirmed from further experiments or ways of physically reducing the rate of reactivity addition need to be found.

Other accidents

20. There was no discussion of accidents involving faulty ECCS initiation, gas ingress from the balloon ECCS to the core and group rod withdrawal, except that it was stated that the group rod case would be easier to detect and compensate for because of the larger perturbation.

I.8.3. Plant specific status

Not applicable.

I.8.4. Recommendations

1. From the reported information it appears that the modifications made to the CPS since the Chernobyl accident have resulted in substantial improvements to RBMK safety. The scope of the accident analysis has been expanded and some western codes have been introduced in calculation of both neutronic and thermal-hydraulic characteristics. However, the margins for the UDBA remain tight and uncertainties in the calculation, particularly of the void effect, are still significant. With 2.4% enriched fuel and other proposed measures, the margins will be larger; however, they have to be compared with typical western margins using common methodology.

2. Selected results of the safety studies performed and referenced in the supporting documentation prepared by RDIPE should be made available for a peer review.

3. An integrated review of both the neutronic and thermal-hydraulic studies is needed since there is significant interaction between them.

4. Independent calculations should be undertaken, using typical safety analysis assumptions applicable in other Member States in order to asses the margins which would normally be expected using such assumptions.

5. Further experiments are needed to understand the blowdown characteristics of the UDBA and its effect on reactivity addition.

6. Western specialists would like in the future to examine the full scope of the accident analysis to ensure that all breaks in piping are covered. Accidents which cause the largest reactivity addition are not necessarily the ones which lead to the largest radioactive releases.

7. The "modernization" of the CPS should proceed expeditiously since the reported improvements would provide considerably better coverage of rod withdrawal accidents and for single channel failures when fast response detectors are installed.

8. Ways should be sought to increase the effectiveness of fast acting shutoff rods to enhance the safety margins for reactivity induced accidents. An optimization study is necessary.

9. Ways should be examined to reduce the amount of reactivity which can be added during failure of the CPS cooling system.

10. Further studies should be undertaken to minimize the effect of the ORM on the void coefficient. Ideally it would be desirable to remove all restrictions but practically ways should be sought to reduce the reliance on administrative procedures in such a critical area.

11. The possibility of thermal-hydraulically induced instabilities for the RBMK should be further discussed and the interaction with both neutronics and the control system examined for possible undesirable effects.

Appendix II
PRESSURE BOUNDARY INTEGRITY

II.1. DESIGN OF THE FUEL CHANNEL

Scope of review:	Fuel channel components; Operation of fuelling machine on channels; Interaction of graphite with fuel channels; Service experience and failures; Inspection of fuel channels; Hot cell testing of irradiated pressure tubes; Replacement of fuel channels.
Document/section reviewed:	[6]

II.1.1. Summary of discussions

The fuel channel tubes are made of Zr 2.5 Nb alloy diffusion welded to Ti stabilized stainless steel end-pieces. The Zr 2.5 Nb pressure tubes are 80 mm in diameter with 4 mm wall thickness and lateral location planes 9 m apart. Provision is made for axial expansion of the tubes relative to the lateral location planes, i.e. the upper stainless end-piece is fixed (welded to the structure) while the lower end-piece connection allows for axial movement. The related discussion is summarized below.

RBMK specialists described fuel channel design and continuing efforts to monitor the integrity of channels. Some 1600 of the total of 2000 channels are fuelled, with the remaining channels used for control rods, cooling graphite and other purposes. The fuel channel schematic drawing is given in Fig. 4.

Three RBMK fuel channels have ruptured due to overheating, which was caused by lack of coolant flow. One break was described as 1 m long with a 120 degree opening. There were no damage, leaks or breaks of adjacent fuel channels. However, movement of graphite, which resulted in deformation of the adjacent channels, was the reason to remove fuel from some of the adjacent channels. The pressure tubes concerned were not replaced but fuel channels affected were taken out of further service.

Some 46 fuel channel tubes have leaked in the dissimilar weld region and have subsequently been replaced. Leakage was from stress corrosion cracks in the stainless steel section of this weld associated with low titanium concentrations.

RBMK specialists discussed hot cell examinations of material from fuel channel tubes removed from service. Measurements have looked at fracture toughness, hydriding, microstructures and corrosion. Some 24 channels have been examined, including randomly removed channels and others that have leaked or burst.

Delayed hydride cracking (DHC) has been observed on the outside diameter of fuel channel tubes. Growth of DHC has been associated with high residual stresses, induced by the former technology of straightening tubes. The technology used at present has overcome this problem (due to the resulting lower residual stresses).

Numerical models of irradiation creep have predicted diametrical growth and closure of the gap between the fuel channel tube and the graphite. These calculations have somewhat under predicted the rate of gap closure observed in service, owing to the insufficiency of initial data.

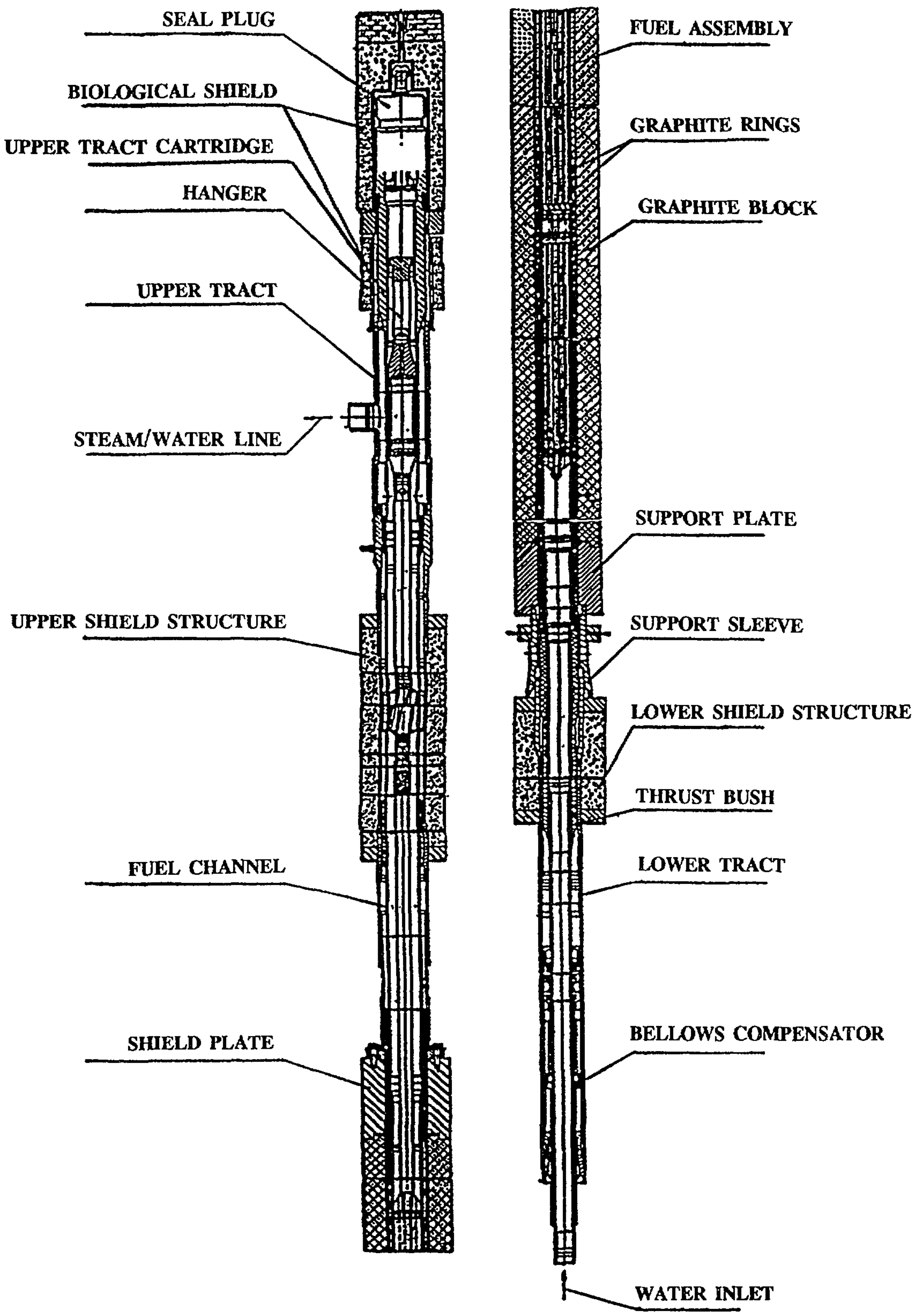
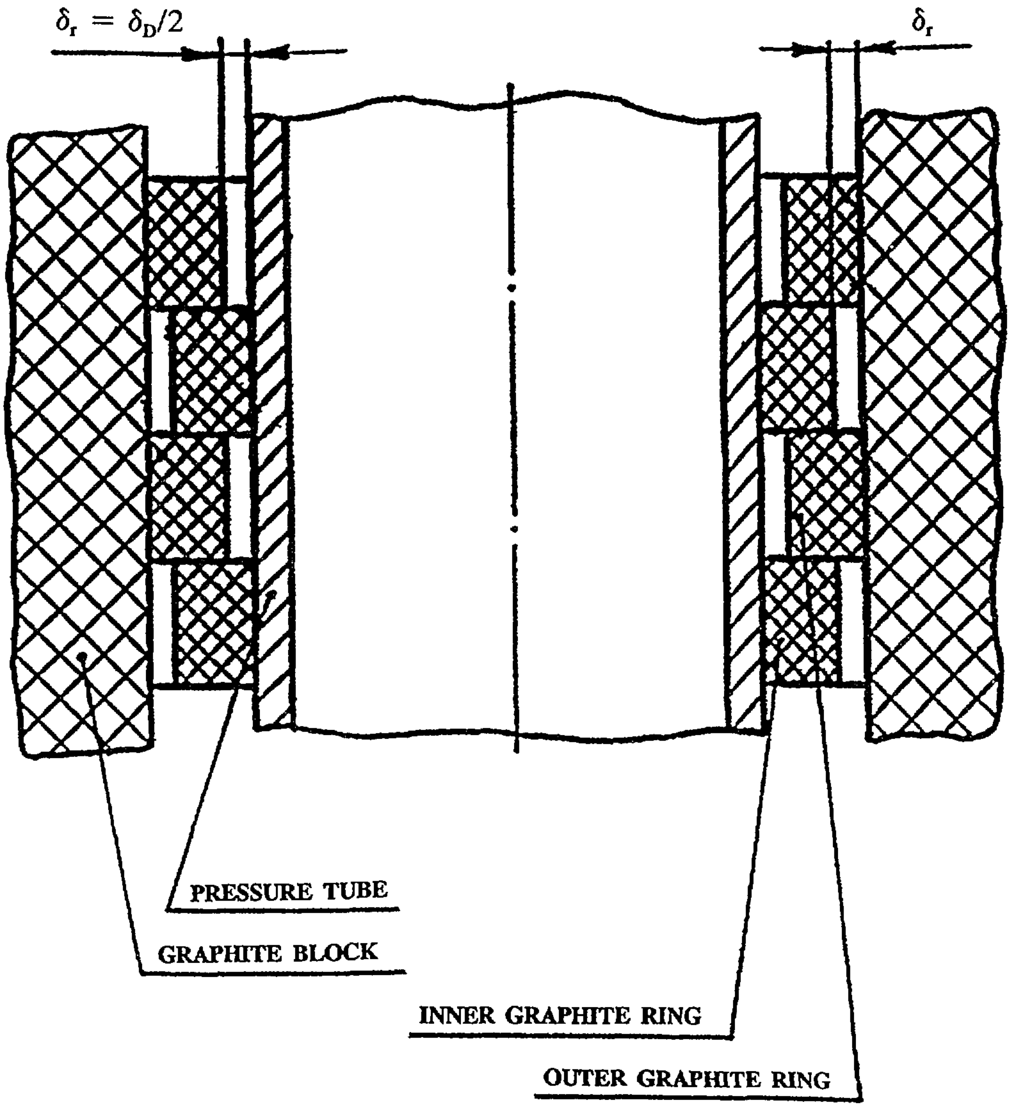


FIG. 4. Fuel channel.



DIAMETRAL GAP δ_D, mm		
INITIAL	BEFORE RETUBING	AFTER RETUBING
2 - 3.46	0-1	2.7 - 4.16

FIG. 5. Schematic of pressure tube and graphite block arrangement of a fuel channel.

TOP

PRESSURE TUBE

GRAPHITE BLOCKS

BOTTOM

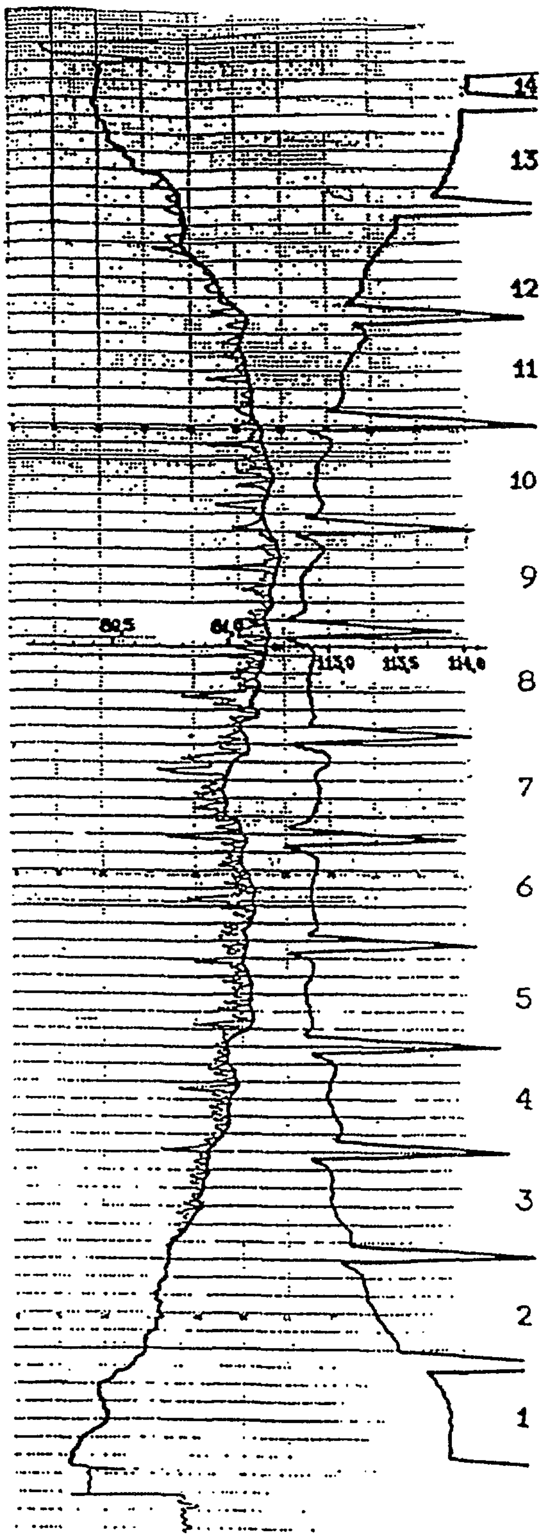
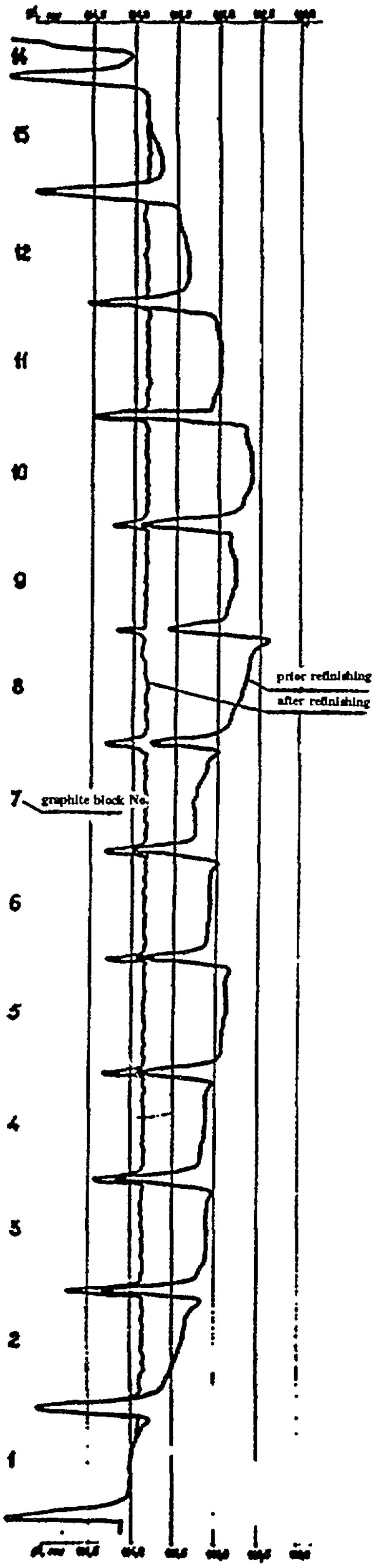


FIG. 6. Profile of fuel channel and graphite blocks inner diameter prior to retubing operation, Leningrad 1, fuel channel 47-44.

TOP



BOTTOM

FIG. 7. Profile of graphite blocks inner diameter prior to and after refinishing.

Replacement of fuel channel tubes is being performed on a routine basis. The Kursk Unit is the next station which will require channel replacement. The gap between the graphite inside diameter and the channel outside diameter is the criterion for replacement rather than loss of structural integrity (see Fig. 5 for the detail of separation between graphite and fuel channel and Fig. 6 for the closure of the gap between graphite and fuel channel after about 15 years of service). Postponement of replacement would preclude future replacement without damaging the graphite. The graphite blocks are bored to enlarge the diameter of the channel passage (Fig. 7 shows the refinished graphite blocks).

Discussions addressed the possibility of unstable failure of fuel channels and analysis by methods using COD (crack opening displacement) parameters and fracture toughness values which are obtained from specimens cut from tubes.

The diffusion bonding of the zirconium to stainless steel was discussed. There has been no history of joint failures.

In-service inspections are performed on fuel channels at all plants using visual, eddy current and ultrasonic methods. Leak monitoring has also been effective in indicating the presence of through wall cracks.

Design loads have included startup/shutdown thermal expansion, thermal transients, electric power interruption accidents, seismic (on a generic basis) and emergency cool down. The maximum design accident is due to rupture of the pressure header (900 mm diameter).

A 5 x 5 full scale mock-up of cells has been constructed to study the effects of a tube burst on adjacent tubes. The actual conduct of the test has been postponed pending action by the Ukrainian participants. The facility is in the Ukraine. The facility was built before the dissolution of the Soviet Union.

There was an extended discussion of the refuelling operation and its effect on fuel channels. There have been some minor leaks of closure plugs, but no cases where a closure plug has been lost. On a few occasions, the refuelling machine itself has had to be used to temporarily seal the end of the channel until the unit could be shut down. Seismic qualification of the refuelling machine was the subject of a lengthy discussion.

RBMK specialists confirmed that there was no essential difference among the fuel channels of the various RBMK reactors. On the 1500 MW(e) reactor (Ignalina) different heat treatment procedures were used for the pressure tubes (for each unit a different heat treatment).

II.1.2. Findings

1. Only three incidents of ruptured channels have occurred during operation but no beyond design basis accidents have occurred to date (no leaks in adjacent channels).

2. There have been 73 cases of detectable leakage resulting from through wall cracks smaller than critical size, exhibiting leak before break behaviour.

3. All stations perform inspections of fuel channels using visual, eddy current and ultrasonic methods and measure dimensional changes on a regular basis during annual maintenance outages.

4. Tubes are being extracted from reactors for hot cell studies of mechanical properties, hydriding, fracture properties, corrosion state and microstructural studies.

5. In some RBMK units all fuel channels have been replaced because of closure of gaps between the tube and graphite due to creep of the tubes and shrinkage of the graphite. The group agreed that the study of creep phenomena should be continued.

6. The diffusion weld between zirconium and stainless steel was discussed and experience has shown this joint to be problem free during thermal cycling of the plant.

7. Operation of the refuelling machine was also reviewed in detail, including the methods of sealing both the channel closure and the refuelling machine. Seismic qualification of channels and the refuelling machine were discussed, and a suggestion was made that seismic qualification might be peer reviewed by an IAEA team.

II.1.3. Plant specific status

See Table II.

II.1.4. Recommendations

1. Volumetric inspections of the channels would be useful to analyse the development of subcritical cracks. An eddy current type of device carried by the fuelling machine may enable this inspection to be done with each fuel change.

2. There should be no reduction in the present program of removing tubes and examining them in hot cells, determining the physical properties the changes in creep, and the dissolved hydrogen, as well as any formation of hydrides. On tube removal the graphite shrinkage should be measured and precise methods of gap prediction should be developed on the basis of these and the tube measurements.

3. The inspection program should incorporate material relationships to identify locations of channels with higher probability of stress corrosion cracks. Inspections should then be planned on the basis of the results of these predictions.

4. The plans to perform tests on the 5 by 5 channel mock-up facility located in the Ukraine should be implemented. Attention should be given to the effects of defective tubes adjacent to break location. Sufficient instrumentation should be used to enable measurement of all the parameters which will allow calibration of the models and validation of the numerical model results.

5. A program to correct problems with flow control valves at the inlets to the channels should be fully implemented. Fail safe characteristics of replacement valves should be fully verified. Modifications should be considered to allow non-operator-initiated response to flow restrictions. (See also recommendations for the primary circuit.)

II.2. REACTOR COOLANT SYSTEM DESIGN

Scope of review:	Primary circuit; Steam and feedwater circuit; Emergency core cooling system (partial).
Document/section reviewed:	[7]

II.2.1. Summary of discussions

The discussion was initiated by presentation of a description of the RBMK primary system by RBMK specialists. The designers consider the primary circuit to be from the steam drum through the pump to the bottom of the channel and then through the channel back to the steam drum. The steam separator, however, is a common point with the steam and feedwater circuit and therefore this circuit is also reviewed. The presentation was as described in the reference document; of special interest however were the following points outlined below.

The circuit is made of austenitic stainless steel piping and components, or of carbon steel clad with austenitic stainless steel, with the exception of some valve bodies which are made of carbon steel and the pump bowls which are made of low alloy chromium steel.

The sources of primary circuit material have been in Russia, France and Japan, and some of the separators have been made in Yugoslavia as well as in Russia. The use of steels from different sources was not due to any technical considerations but rather to commercial and other considerations. Supply from Yugoslavia was as a result of an agreement on industrial technical transfer.

The steam separator has more than 400 nozzle connections. Some concern was expressed about the detail of the nozzle design; a sketch showing how these nozzles are manufactured in such large numbers on the stainless steel clad vessel is attached (Figs 8 - 11). The need for this detail was reinforced by the fact that on WWER reactors nozzle connections of stainless steel piping to clad carbon steel vessels have been of concern. The approach to this type of two material connection was to have the transition made at the factory, and to leave a simple stainless to stainless pipe weld for the field joint. To date there have been no cases of a connection failure.

Analysis has shown that the total failure of either the inlet or the outlet tube of the channel still allows reliable cooling of the fuel. Partial failures of these tubes and the fluid stagnation is being studied by the accident mitigation specialists.

Some discussion was carried out on the seismic qualification of the channel inlet and outlet tubing. These tubes are seismically qualified. Until 1986 the equipment was analysed to common seismic characteristics; now seismic parameters are considered on a site specific basis. Site specific seismic analysis for Leningrad nuclear power plant is complete while for the other plants the work is still in progress. A typical ground level response spectrum is attached to this report to indicate to which severity they have been qualified (see Figs 12 and 13). The tubes have also been vibration tested and as a result some additional supports have been added.

A horizontal spacing of 250 mm and a vertical spacing of 90 mm is maintained on the hot outlet tubes.

The pipe whip of the feeder piping was discussed. It was stated by RBMK specialists that calculations indicated that pipe whip forces of a ruptured tube do not rupture adjacent tubes and that experiments had shown that the forces may deform tubes but they would not rupture. Similar experiments were conducted in Japan and led to the same conclusions.

In 180 reactor years of operation with about 1600 lines per plant, there have been some instances of through wall pipe cracks. Two have been analysed to be vibration induced cracks, others were manufacturing or assembly induced effects. The recent (6.10.1992) leak at the Ignalina plant at an elbow was due to vibration. The leakage was detected by radiation monitoring. The crack length was 12 mm. The crack could not be found after the plant was shut down until a pressure of 85 kg/cm^2 was applied. The leakage rate measured on the basis of a known level of activity in the water was 300 L/h. There was also a case where cold water dripping onto a hot tube induced a crack. This event occurred about ten years ago and the cause for this possibility has been eliminated by the addition of a screen between the hot and cold water lines.

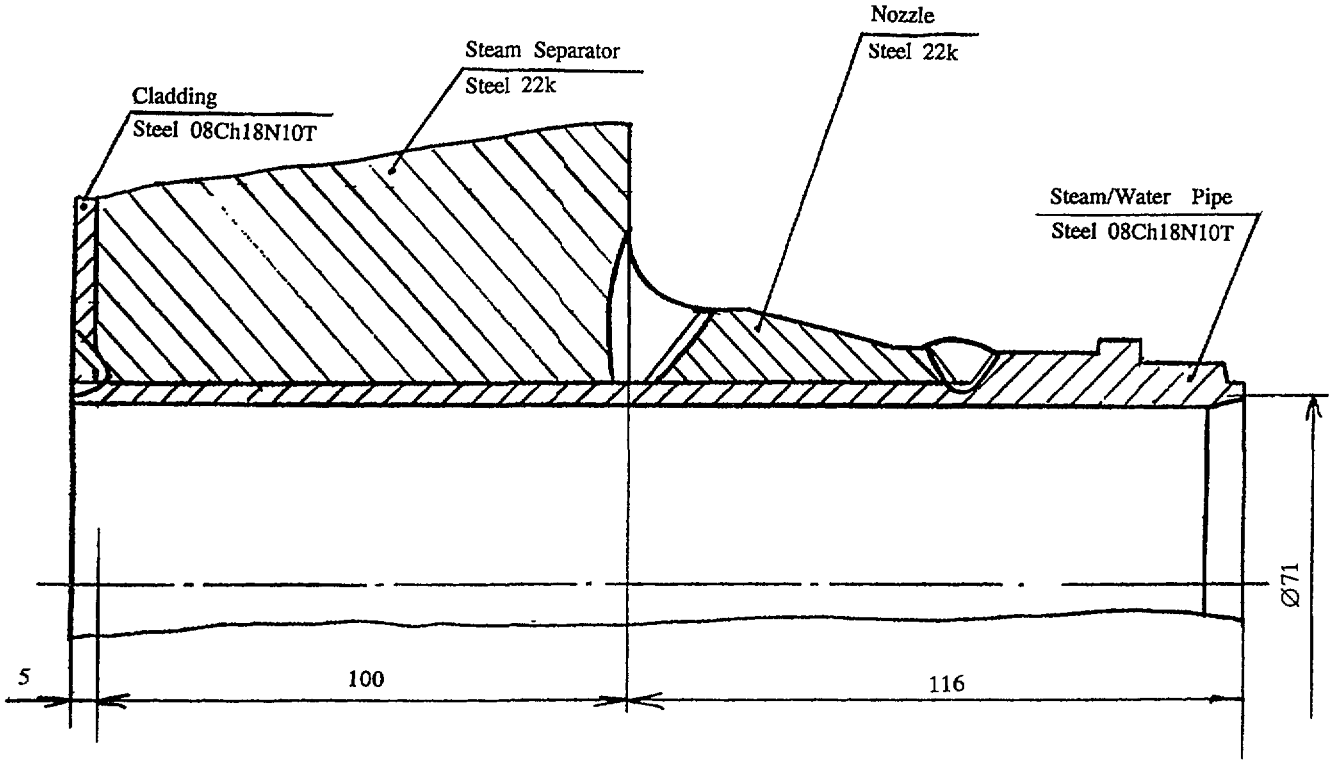


FIG. 8. Reactor coolant system nozzles: steam/water line - steam separator.

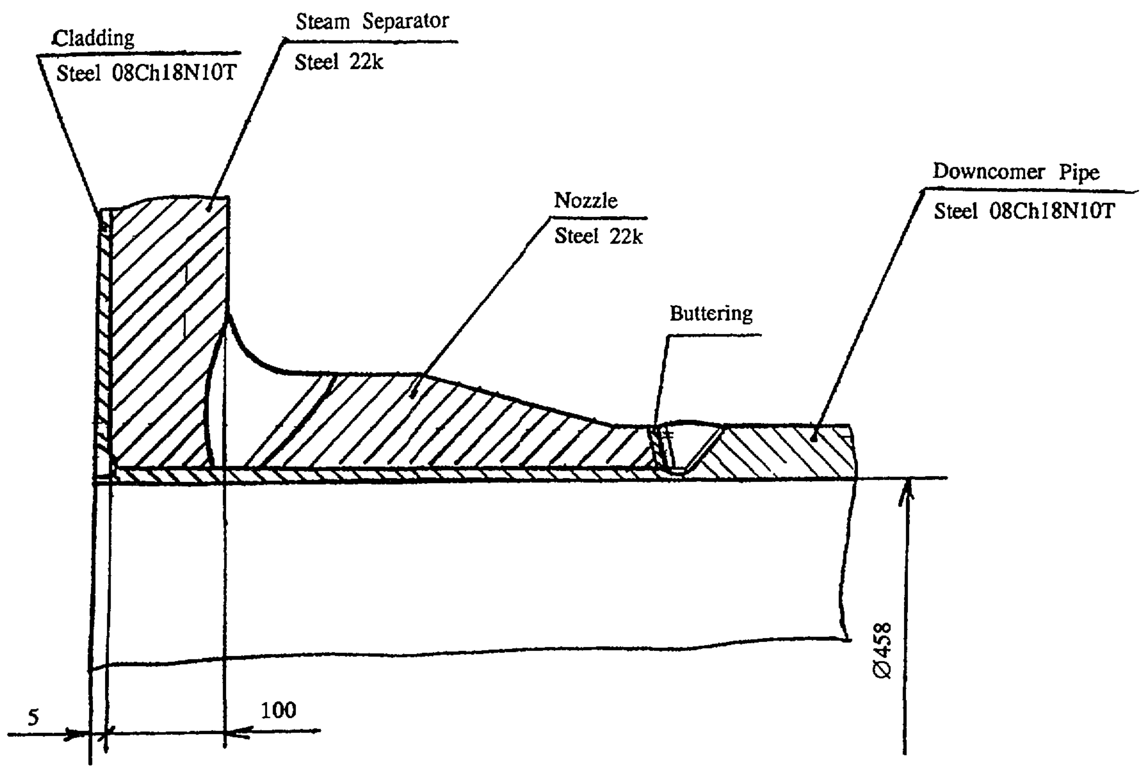


FIG. 9. Reactor coolant system nozzles: steam separator - downcomer.

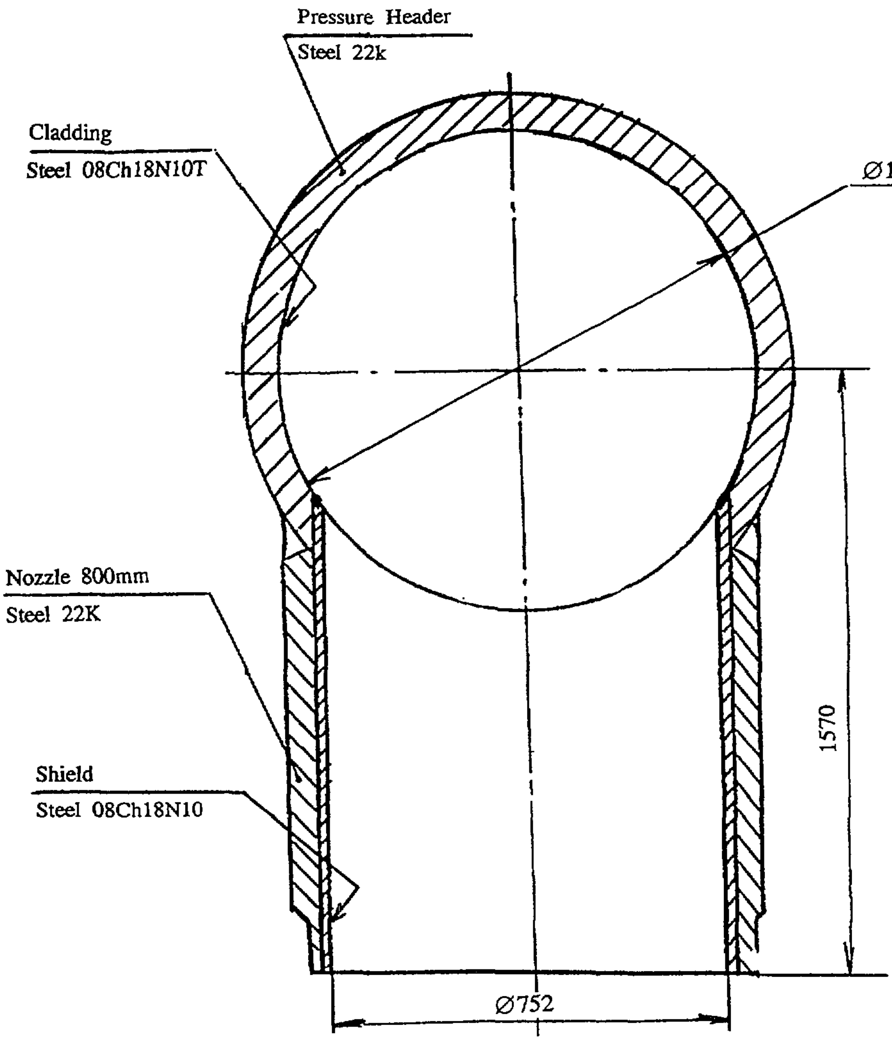


FIG. 10. Reactor coolant system nozzles: feeder line - pressure header.

In response to a question as to the occurrence of erosion on elbows, it was confirmed by the RBMK specialists that the elbows are monitored for thickness and that to date no thinning has been seen on the feed water inlet or outlet piping.

Inspection of the system is carried out visually, by dye penetrant, radiography and ultrasonic methods. Each reactor unit has about 58 km of 68 mm stainless steel piping upstream of the core and 40 km of similar diameter piping downstream of the core. Each year the piping is visually inspected and 5 + 5 (upper and lower parts) welded joints are volumetrically examined. Larger lines and the separator vessel are examined every four years. Automatic inspection equipment for headers and other larger components is now being developed. The first results obtained by the new automatic equipment are compared with the finding of the Finnish specialists for the same components. New imaging techniques are being used to display the data. The Leningrad nuclear power plant is being used as a pilot plant for this work. Supports of equipment and piping are visually inspected starting after 15 000 hours of operation and then every 45 000 hours. They are also inspected after any unusual transients.

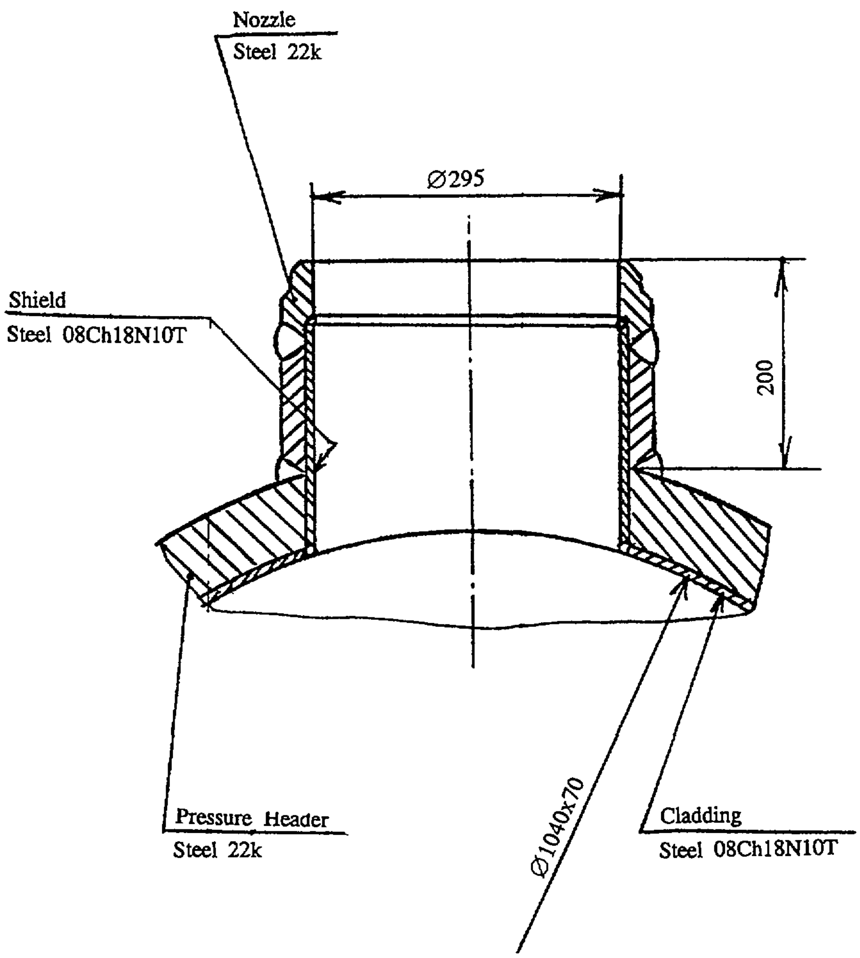


FIG. 11. Reactor coolant system nozzles: pressure header - distribution group header.

In response to a question on the support of the primary system it was established that the system is supported as follows:

- top of fuel channel fixed;
- outlet 68 mm piping fixed only in transverse direction;
- steam separator fixed horizontally at centre support and not restrained from upward movement;
- no meaningful differences in support from plant to plant;
- no problems recorded resulting from the support scheme for the primary system.

Stress corrosion is controlled by control of the coolant water chemistry. Drains have been added to stagnant zones in the circuit to make the water more effective. In the circuit there are containers where both stressed and unstressed samples may be placed, exposed to the coolant, and later retrieved for study to measure the effectiveness of the water chemistry control. Every 100 000 hours of operation, samples of material are cut out of the circuit to allow exact study of the situation. In the past acid was used for decontamination of the circuit but was found to corrode the carbon steel component sections and the practice has been discontinued. The zirconium/2.5% niobium fuel channels are protected by autoclaving them to obtain a hard black oxide film. After some time of operation some losses of this film have been noted. The causes are not known but they may be associated with the fact that the fluid in that section is mainly steam with some water which may cause erosion and/or fuel vibration.

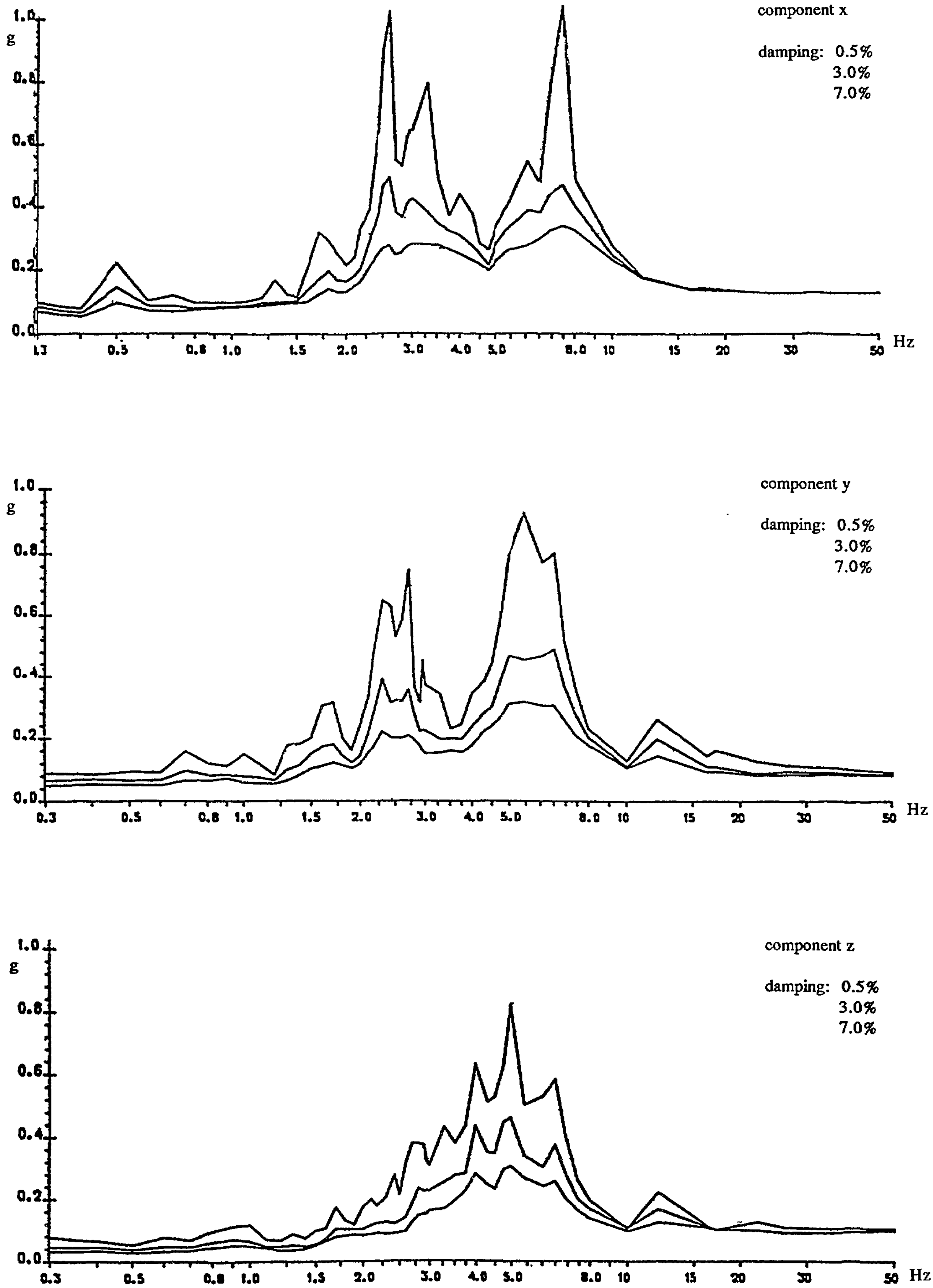


FIG. 12. A typical ground response spectrum at the foundation level. Soil deformation module 300-400 kg/cm², (centre of the base mat).

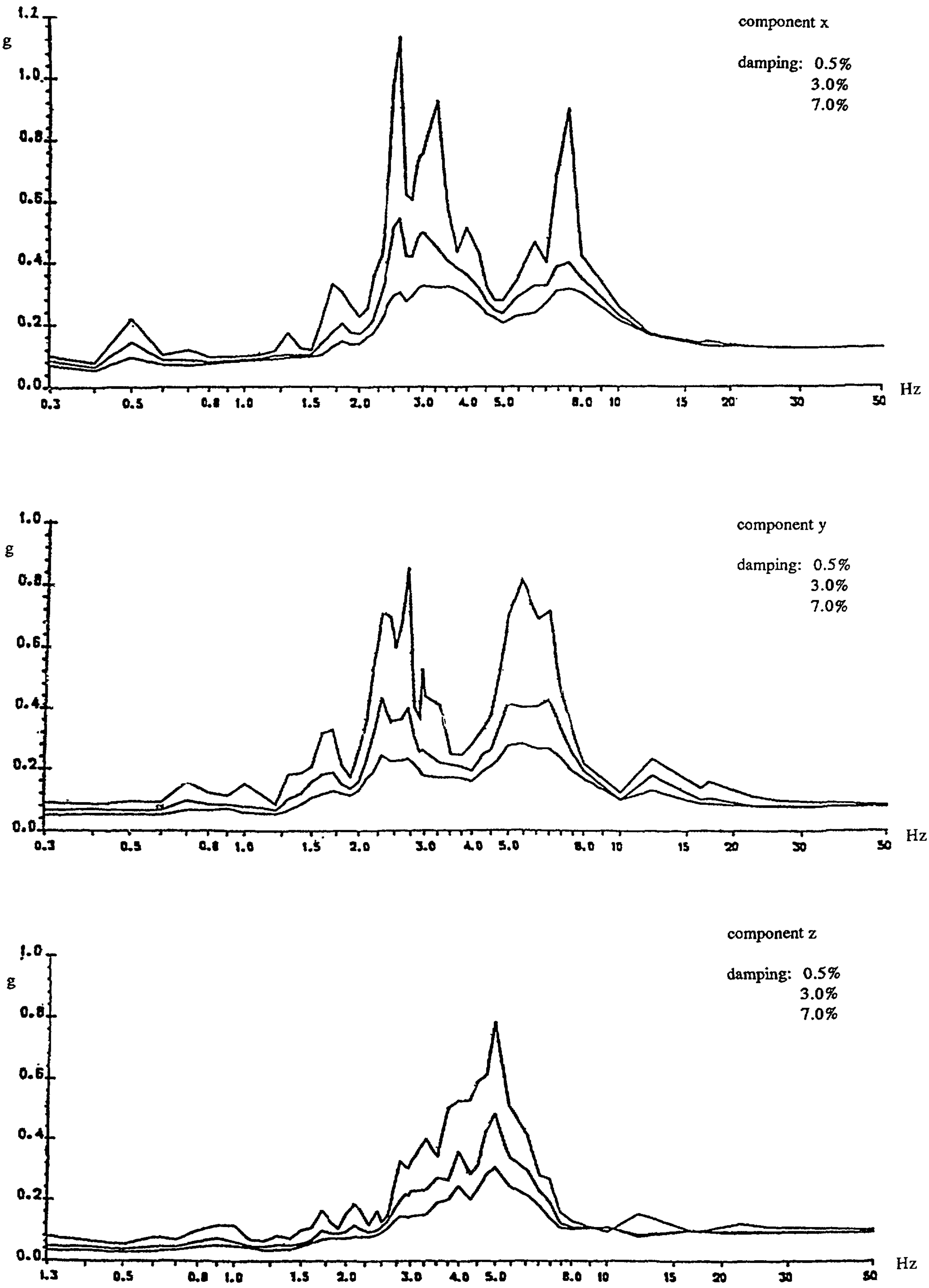


FIG. 13. A typical response spectrum at the reactor foundation. Soil deformation module 300-400 kg/cm².

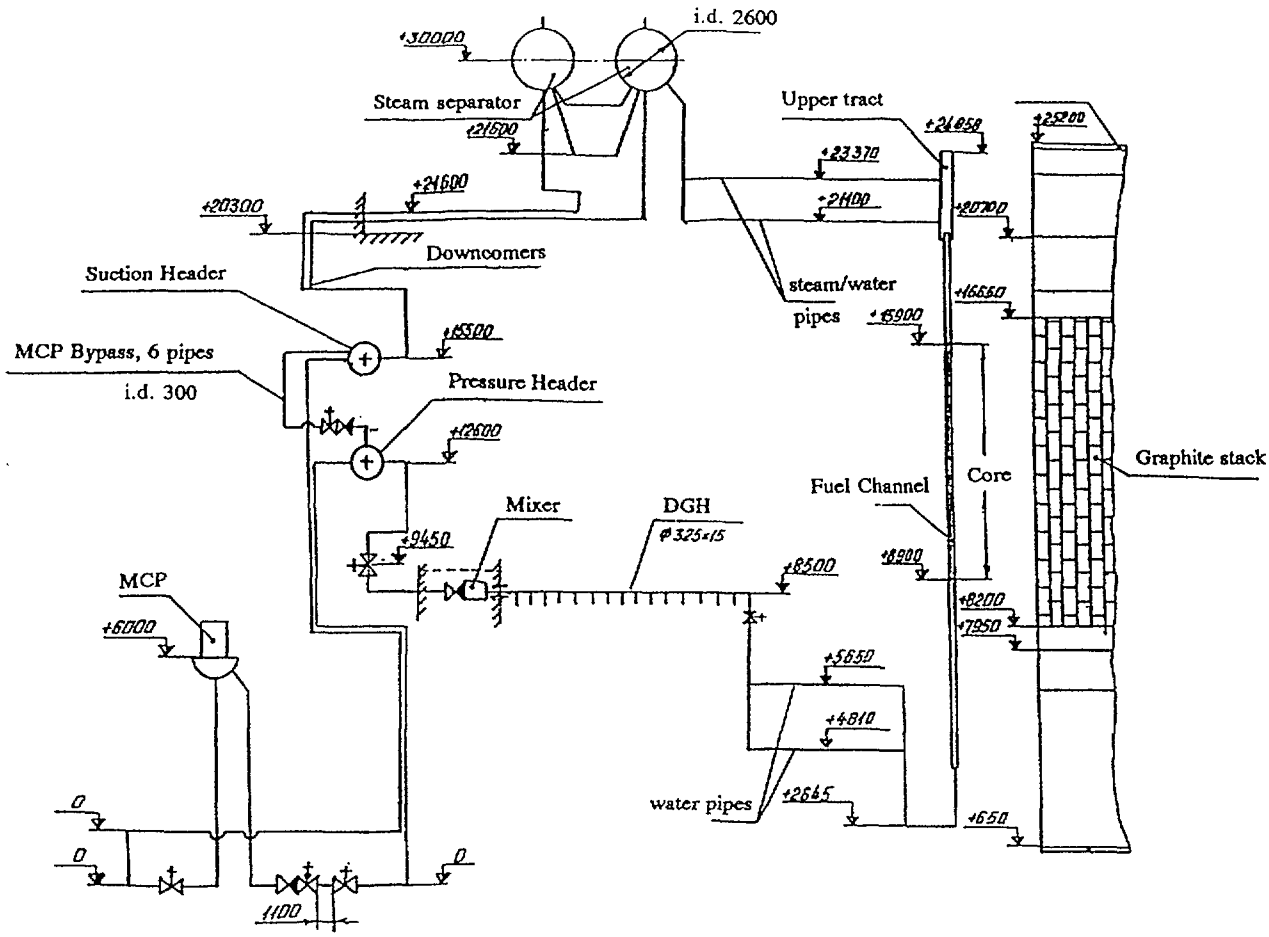


FIG. 14. Main circulation circuit flow diagram.

TABLE IX. STEAM SEPARATORS

Unit	Diameter	Material	Manufacturer
Chernobyl 1	2300	22K	Izhora
Chernobyl 2	2300	22K	Izhora
Chernobyl 3	2600	Creuselso 330E	Yugoslavia
Kursk 1	2300	22K	Izhora
Kursk 2	2300	22K	Izhora
Kursk 3	2600	Creuselso 330E	Yugoslavia
Kursk 4	2600	Creuselso 330E	Yugoslavia
Leningrad 1	2300	22K	Izhora
Leningrad 2	2300	22K	Izhora
Leningrad 3	2600	22K	Izhora
Leningrad 4	2600	22K	Izhora
Smolensk 1	2600	Creuselso 330E	Yugoslavia
Smolensk 2	2600	Creuselso 330E	Yugoslavia
Smolensk 3	2600	Creuselso 330E	Yugoslavia
Ignalina 1	2600	22K	Izhora
Ignalina 2	2600	22K	Izhora

TABLE X. SUCTION (DIAMETER 900 mm) AND DISCHARGE (DIAMETER 900 mm) HEADERS

Unit	Material component	Layout
Chernobyl 1	1	A
Chernobyl 2	1	A
Chernobyl 3	2	B
Kursk 1	1	A
Kursk 2	1	A
Kursk 3	2	B
Kursk 4	2	B
Leningrad 1	1	A
Leningrad 2	1	A
Leningrad 3	3	C
Leningrad 4	3	C
Smolensk 1	2	B
Smolensk 2	2	B
Smolensk 3	2	B
Ignalina 1	3	D
Ignalina 2	3	D

Key:

1 - Steel (22K) and component manufactured at the Izhora plant.
2 - Steel (Creuselso 330 E) manufactured at Creusot, France; component manufactured in the former Yugoslavia.
3 - Steel (19MN5) and semi-products (pipes) manufactured in Japan, component manufactured at the Izhora plant.
A, B, C, D: Different layout of components in the section from the drum separator to the distribution group header.

TABLE XI. CONFIGURATION OF DISTRIBUTION GROUP HEADERS

Unit	Check valve on the inlet of the distribution group header	Gate valve on the inlet of the distribution group header	Number of distribution group headers
Chernobyl 1	no	yes	44
Chernobyl 2	no	yes	44
Chernobyl 3	yes	yes	44
Kursk 1	planned for 1993	yes	44
Kursk 2	planned for 1996	yes	44
Kursk 3	yes	yes	44
Kursk 4	yes	yes	44
Leningrad 1	yes	no	44
Leningrad 2	yes	no	44
Leningrad 3	yes	no	44
Leningrad 4	yes	no	44
Smolensk 1	yes	yes	44
Smolensk 2	yes	yes	44
Smolensk 3	yes	yes	44
Ignalina 1	yes	yes	40
Ignalina 2	yes	yes	40

TABLE XII. CONTROL/ISOLATION VALVE ON FUEL CHANNEL INLET

Unit	Valve	Layout
Chernobyl 1	2	A
Chernobyl 2	-	A
Chernobyl 3	1	B
Kursk 1	1	A
Kursk 2	2	A
Kursk 3	3	B
Kursk 4	3	B
Leningrad 1	1	A
Leningrad 2	1	A
Leningrad 3	1	C
Leningrad 4	1	C
Smolensk 1	3	B
Smolensk 2	3	B
Smolensk 3	1	B
Ignalina 1	2	D
Ignalina 2	1	D

Key:
The valves of new design were built in to all plants put into operation from 1987 and onwards.
1 - The new design valve has been installed. Unit is in operation.
2 - The new design valve has been installed. Unit is shut down and being repaired.
3 - The old design valve is installed. The minimum valve opening is two times larger to avoid the recurrence of accidents like that in Leningrad 3 in March 1992. The new design valve will be installed in 1993.
A, B, C- Different layout of components in the section from steam separator to the distribution group header.

Information on the steam and feedwater circuit was presented by the RBMK specialists. This circuit which is made of carbon steel is designed by the station designer (Atomenergoprojekt, except Ignalina which was designed by VNIPIET). A rough flow sheet is attached to the report based on the Ignalina plant (see Fig. 14 for the first circuit flowsheet, Fig. 3 for the steam circuit flowsheet and Fig. 2 for a cross-section of the reactor building). The plant specific details were also discussed (see Tables IX-XII).

The basic arrangement is that one reactor drives two turbines, each rated at approximately one half of the reactor output. All plants have both main steam isolation valves, which close in about two minutes, and turbine stop valves which close rapidly (in 0.4 s). There is a series of valves which allow about half the steam to be bypassed directly into the condenser. There are also emergency bubbler type condensers on the first generation reactors and a suppression pool which uses the same principle on the second generation reactors.

There have been fewer than 10 cases of through wall cracks in the water levelling pipes of the steam separators.

II.2.2. Findings

1. The main circuit may be divided into the part of the circuit which flows through the channels and the part of the circuit which flows from the separator to the steam turbine and then

through the feedwater system back to the steam separator. The first circuit interface with coolant is mainly stainless steel, although there are some carbon steel valves; the second circuit is mainly carbon steel.

2. Each feeder pipe has a flow control valve to the channel inlet side and the flow is periodically manually adjusted, first when the fuel is replaced and then as the fuel is depleted and the contribution of the channel decreases to the reactor energy output.

3. Periodic inspections are performed on the bases of specific numbers of operating hours. Some of these inspections are visual and their efficiency seems dubious.

II.2.3. Plant specific status

The plant specific status is given in Tables IX-XII.

II.2.4. Recommendations

1. The possibility of backfitting better containment-confinement systems should be studied.

2. A peer review should be carried out on the reliability analysis of the first and second circuit to ensure that the system of valves and the various interconnected safety systems are rational and will perform with the appropriate reliability the function for which they have been designed.

3. Additional safety analysis may be required at the plants where the safety improvements such as installation of check valves at distribution headers, etc., have not yet been made.

4. The possibility of using motorized valve operators and direct computer control for functions such as valve regulation should be investigated.

5. The design of the nozzles and joints in the main circulation circuit (in particular those containing dissimilar welds) present concerns as to their structural integrity and potential for cracking, especially in the latter part of their service life. The connection design should be further studied for the effects of ageing and the development both of corrosion cracking and carbon steel/stainless steel separation.

6. The water chemistry of the reactor and steam circuit is of primary importance since different materials are used in the circuit, e.g. carbon steel,stainless steel and zirconium-niobium alloy. New methods of decontamination of the circuit should be developed to reduce exposure to the in-service inspection personnel without eroding the material of the circuit.

7. Site seismic analyses are now in progress or completed for all reactors, and these will augment the technical basis provided by the generic analysis. These site specific analyses should be peer reviewed, and any indications of equipment or support weaknesses corrected and backfitted. Of special interest is the support of the refuelling machine, which should be analysed to maintain a condition of remaining locked on to a channel during a seismic event without leakage, or any damage to the channel or the machine.

8. The system periodic inspection program should be optimized. This is discussed further under the subject in Appendix II.3.

II.3. REACTOR COOLANT SYSTEM INSPECTION

Scope of review:	Methods, scope and frequency of inspection; Reliability of inspection methods; Laboratory examination of surveillance samples; Rejection criteria for degraded material.
Document/section reviewed:	[8]

II.3.1. Summary of discussions

RBMK specialists made a presentation to describe inspection programmes (Table XIII) that are all guided by a single regulatory document. This document was prepared by the RBMK Design Institute and has been adopted by regulatory authorities. Inspections include the following:

- all the vessel welds of the steam separators excluding penetration welds are inspected once every 4 years by UT;
- nozzle welds are inspected under other rules, 5 nozzles per separator once every 4 years;
- large diameter ferritic piping has all welds inspected by manual UT once every 4 years;
- 300 mm stainless piping has welds inspected by radiography;
- UT calibration is performed on side drilled holes;
- valves undergo 100% visual inspection;
- ten elbows of each size of the steam pipe lines are inspected by UT and MT (magnetic particle).

Inspections are performed by plant personnel at each site. Special institute provides training and certificates that qualify people. Limited evaluations of performance of field inspectors have been conducted by researchers taking samples with artificially induced cracks to check performance of plant personnel.

Inspection optimization has been discussed. RBMK specialists stated that RBMKs lack containment and this has motivated 100% inspections of some components. Inspection focusing on those components that have the greatest contributions to risk and higher quality NDE on these components have been proposed.

The exposure of metal surveillance samples to reactor environment (radiation and corrosion) was discussed. The impact of the examination of these specimens was also discussed. Heat treatments of zirconium tubes have been modified based on surveillance trends.

Inspections of fuel channels in RBMKs, in Canadian CANDU reactors, and in Japan were discussed. It was stated that UT inspections in RBMKs were first performed in response to fuel channel leakage observed in 1982 (visual inspections being performed prior to that year) and are required at all plants by regulation since then. The present status of fuel channels inspection is given in Table XIII. The new regulation covering the in-service inspection of fuel channels and control and protection system channels, a draft of which is being reviewed, is to be put into force from 1 January 1993 pending final approval. It specifies the scope and volume of inspection; however, the volume is limited to a maximum of 50 of the approximately 1600 channels (Table XIV). In contrast, the CANDU pressure tubes are inspected on a 100% sample basis, and 10% of tubes in Japan are inspected once every 10 years.

RBMK channels are now being replaced as needed due to gap closure by creep. Other less restrictive replacement criteria are degradation of mechanical properties and deformation of the

TABLE XIII. FUEL CHANNEL INSPECTION (Information on three RBMK units)

Unit	Visual inspection of fuel channels						Measurement of fuel channel diameter						Ultrasonic inspection of fuel channel						Measurement of the graphite hole diameter					
Year	88	89	90	91	92	93	88	89	90	91	92	93	88	89	90	91	92	93	88	89	90	91	92	93
Chernobyl 1																								
Chernobyl 2																								
Chernobyl 3																								
Kursk 1		15		13					21	13					10					3	10			
Kursk 2																								
Kursk 3																								
Kursk 4																								
Leningrad 1																								
Leningrad 2																								
Leningrad 3																								
Leningrad 4																								
Smolensk 1																								
Smolensk 2																								
Smolensk 3					20																			
Ignalina 1	9		10	30			26	22	20	30					34	20								
Ignalina 2																								

TABLE XIV. VOLUME AND SCOPE OF PROPOSED REGULATION FOR FUEL CHANNEL INSPECTION

Method	Number of Fuel Channels inspected in given steps 1-4				Comment
	I	II	III	IV	
Visual	50	10	30*	30*	Consequently all channels in this group are tested
UT	50	10	30	30	Consequently all channels in this group are tested
Measurements of:					
- diameter	50	10	30	30	Consequently all channels in this group are tested
- wall thickness	50	-	-	15	Consequently all channels of this group are tested. Thickness assessment
- Zr2.5Nb tube length	50	-	-	15	Consequently all channels of this group are testes. Length assessment
- lateral distortion	50	-	-	10	After 20 years of operation inspection volume to be reassessed
Post-reactor hot cell examination	-	1	2	2	-

* Only 15 channels are inspected if decontamination of the main circulation circuit is not performed prior to inspection.

Step I: During reactor assembly or complete re-tubing.

Step II: After first 8 000-10 000 h of operation of channels.

Step III: Periodically each 18 000-20 000 h during the 15 years of operation of channels.

Step IV: Periodically each 8 000-10 000 h after the first 15 years of operation of channels up to reactor retubing or shutdown (decommissioning).

channel. Channels are removed at each plant on a regular basis (beginning at 8000 hours) to permit hot cell measurements of mechanical properties.

Samples of piping materials are exposed to reactor environments in surveillance capsules and subject to corrosion and mechanical property tests. Samples have also been cut from actual components. Data from the Leningrad plant showed no unexpected changes in properties.

The joint of steam/water pipes to the steam drum was discussed (see Fig. 8). This is a difficult metallurgical joint of carbon steel to stainless steel. Although the configuration of this nozzle joint is not covered by the ASME Pressure Vessel Design Code, it has performed well to date at all RBMK plants.

II.3.2. Findings

1. RBMK primary circuit piping and components are inspected for defects in accordance with a Russian regulatory document. Few if any defects have been found as a result of these extensive inspections.

2. Leak detection has been used to monitor both fuel channels and the primary coolant circuit, and the detection methods have been effective in finding leaks in degraded RBMK components.

3. The RBMK organizations have been developing improved methods for ultrasonic inspection of fuel channels and the primary circuit. These methods are slowly being put into practice on a pilot basis.

4. Non-destructive evaluation of fuel channels by UT or other methods is only now being implemented at all plants, and only a small sample of channels will be inspected when it is implemented. Leak monitoring will remain the main method of detecting degraded tubes.

II.3.3. Plant specific status

Inspections are performed on a uniform basis from plant to plant in accordance with a regulatory document. In some cases new inspection technology is applied to specific plants on a pilot basis.

II.3.4. Recommendations

1. The current program of in-service inspection should be continued as a means to assure and maintain confidence in structural integrity.

2. Optimization of inspections by identifying the most risk significant locations (probability and consequences of failure) is supported. This could lead to fewer inspections, but of a higher level of inspection reliability.

TABLE XV. EQUIPMENT NEEDED TO ENHANCE IN-SERVICE INSPECTION CAPABILITIES (As suggested by RBMK Specialists)

Fuel channels and graphite inspection	
1.	Equipment to record roughness of the internal FC surface (replica technique).
2.	Equipment to remove scraps from the FC inner surface (metallographic analysis).
3.	Radiation resistant video equipment to record visual examination of FC and graphite stock.
4.	Radiation resistant endoscope (length 20 m) optical fibre with movable end piece and light.
5.	Stereoscopic equipment for reactor internal inspection (e.g. following removal FC, ...).
Leak detection	
1.	Manual He leak detectors.
2.	Radiation resistant optical leak detectors (heat resistant > 100°C).
3.	Heat resistant humidity sensors.
LBB application (leak detection capabilities)	
1.	Description of codes and approaches to evaluate related acoustic signals.
2.	Methods to detect acoustic leak signals within operational noise, evaluation of signals, parameters.
3.	Methods used to evaluate leak rate based on acoustic signals.

3. The usefulness of visual inspection of the primary circuit is questionable, except to ensure that piping supports are intact. Visual inspections should be augmented by inspections at locations of high failure consequences and/or high failure probability, as identified by service experience, piping stress analyses and/or other considerations.

4. As a long range objective, methods of performing a 100% examination of fuel channels should be sought. This will probably require an added capability to the refuelling machine to perform inspections at the time refuelling is completed.

5. Continued inspections of fuel channels should focus on the diffusion bonded joint between the zirconium and stainless steel. These inspections should monitor stress corrosion cracking of the stainless portion of this joint.

6. The effectiveness of NDE methods should be more extensively established through performance demonstrations on specimens with realistic service type defects. These demonstrations should have a statistical basis of the type used to guide the PISC programme.

7. The nozzle/joining between the feeder tubes (diameter 68 mm) and steam separator should be given operator attention. Inspection methods should be improved and demonstrated on simulated service cracks on nozzle/joint mock-ups.

8. The inspection of the main circulation circuit should be carried out regularly (100% UT every 4 years). Steam pipes between the separators and the turbine stop valves, and the feedwater lines between the separator and the feedwater control valves, should be inspected sufficiently to give the same level of confidence as for the primary system, since there are no isolation valves between them.

9. Improvements on training and personnel qualification for NDE is recommended (ASSET missions, comparison of results through international co-operation, etc.).

10. The Russian specialists prepared a list of equipment which would improve the in-service inspection capabilities (see Table XV). It is recommended that this equipment be acquired as soon as possible.

II.4. LEAK BEFORE BREAK CONCEPT

Scope of review:	Leak before break as applied to RBMK reactors; Fracture mechanics methods; Leak detection methods; Numbers and sizes of flaws in welds; External events - aircraft impact, wind loads, flooding.
Document/section reviewed:	[9]

II.4.1. Summary of discussions

A presentation of fracture mechanics approaches utilized in RBMK reactor analyses was given by RBMK specialists. A simplified engineering approach is used since a complete fracture mechanics database (in particular J_R curves) is not available for RBMK primary circuit materials. This strain intensity factor concept and limit stress approach has been used to develop tables of allowable flaw sizes, which regulate which flaws must be repaired upon discovery during in-service inspection.

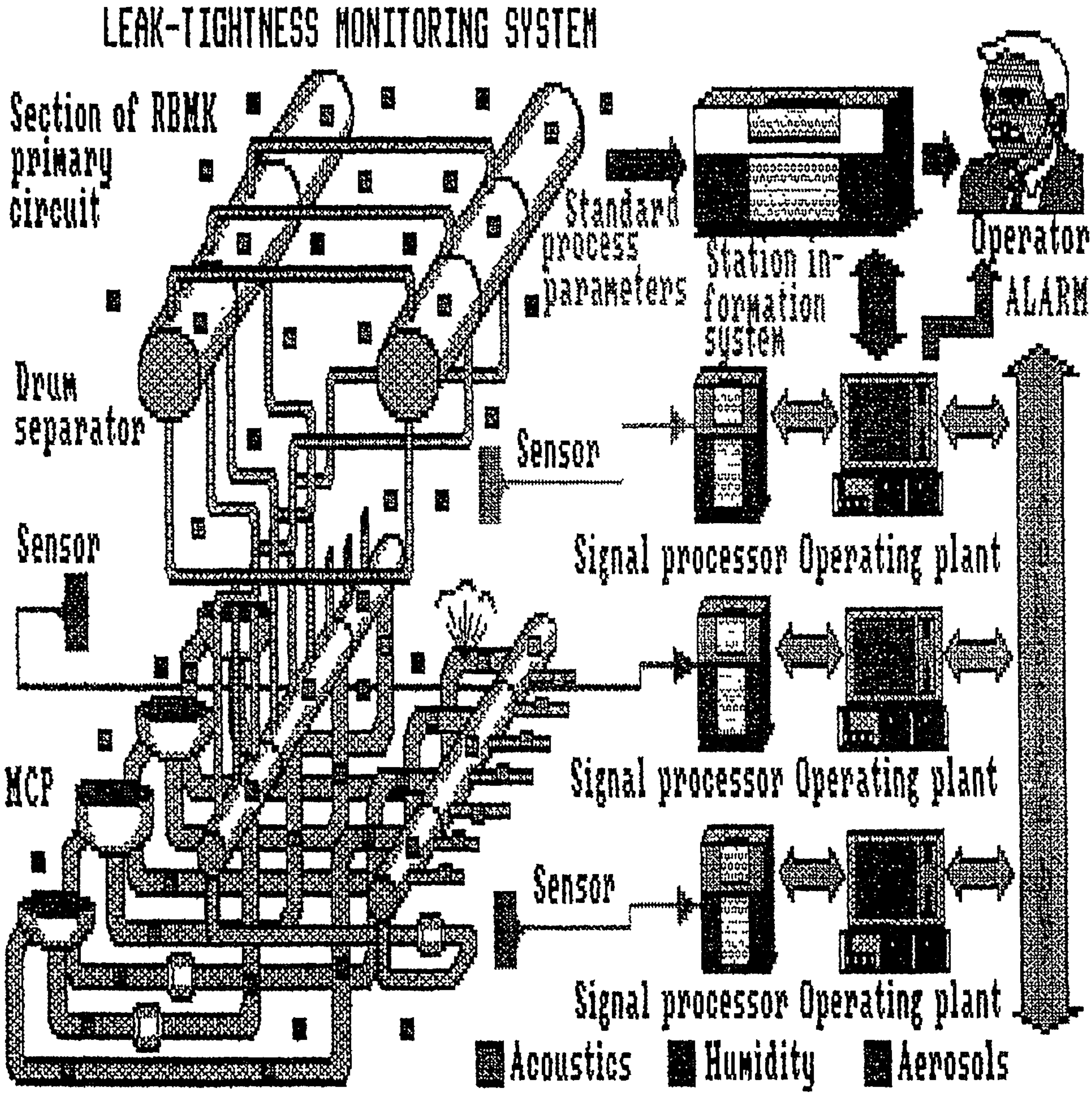


FIG. 15. Leak detection system.

Probabilistic fracture mechanics calculations have been performed for the 800 mm pipe to predict leak and rupture probabilities. Failure of this piping has the highest consequences should a large break occur.

Data on flaw detection probabilities were presented (by ultrasonic inspection) and data on the numbers and sizes of flaws in the welds of 800 mm piping at Ignalina, as well as detected flaws and those believed to remain undetected after inspection. In 400 m of weld a total of 2 to 9 flaws are believed to exist after UT inspection. Such data have been the basis of fracture mechanics calculations, and are a valuable addition to the world database on this subject.

RBMK specialists stated that the leak before break (LBB) was not the technical basis for major RBMK safety concepts, such as containment versus confinement. Analyses of LBB are in a state of development and will be used to properly distribute resources (e.g. leak detection versus ultrasonic detection of cracks).

Leak detection methods at RBMK units were discussed, both for leakage from fuel channels into moderator graphite and from piping into compartments. Detection methods are based on acoustic signals, radiation levels and humidity (Fig. 15). Leakage from fuel channels is detected for leakages of 100 g/h or greater; in other piping, leakages of 50 L/h or greater are said to be detected reliably.

Evaluations of aircraft impact show that this postulated accident can have severe consequences to RBMK units due primarily to fires caused by fuel on board the aircraft.

Evaluations of severe wind loads are performed on a plant specific basis.

Flooding from external sources has been considered at individual plants, including seawater hazards at Leningrad.

II.4.2. Findings

1. The leak before break evaluations are interesting from the viewpoint of fracture mechanics, but have no implications yet on conclusions for reactor modifications.

2. Work is still in progress. Evaluations do not yet prove that the LBB concept applies to the RBMK pressure boundary, but future results could prove valuable in properly allocating resources.

II.4.3. Plant specific status

Leak before break evaluations have been performed on a generic basis, with inspection data from Ignalina used to support evaluations. No modifications to plants have yet been made or proposed based on LBB evaluations.

II.4.4. Recommendations

1. Currently available results related to application of the LBB concept do not provide a basis for decisions on RBMK safety.

2. Provided that the LBB concept is to be applied, plant specific work completed and planned should be analysed and peer reviewed, including stress analysis, fracture analysis, stability analysis, integrity analysis and leak detection.

3. Development of the method of analysis for pipe rupture, including zirconium, stainless steel, and carbon steel materials, taking into account earthquakes, should be continued. The probability studies and data collection should be conducted for each specific equipment layout and meaningful plant difference. An expert group should be formed to do this work.

4. If the leak before break concept is to be applied to RBMK plants, the leak detection system should be improved to detect smaller leaks with high reliability and should be properly operated. First generation RBMK plants do not have a pressure confinement to endure a pipe break for a pipe over 300 mm in diameter.

5. A peer review should be carried out on the steam line leak detection systems, as well as on the safety and isolation valve arrangements.

Appendix III
ACCIDENT MITIGATION

III.1. ECCS DESCRIPTION

Scope of review:	Design of the ECCS Safety Injection for Kursk 1, Smolensk 3, Ignalina 2 and Leningrad.
Document/section reviewed:	[10], [11] Main measures plan to modernize NPPs; Ignalina flow diagram; Smolensk schemes.

III.1.1. Summary of discussions

General

The safety injection is designed to provide cooling of the reactor in the event of primary circuit breaks (inlet cooling) and after the hydroaccumulators have dumped their water into the core. The safety injection will also be used to mitigate the consequences of steam line breaks, but this will be discussed separately.

The most limiting pipe breaks for which the system is designed are the following:

(a) Break of piping or headers of the coolant system in the steam separator compartments, lower water lines room, or leaktight compartments.
(b) Rupture of a distributing group header (DGH), with or without failure of the adjacent check valve.

The maximum diameter of pipe rupture is estimated 300 mm in Kursk 1 (1st generation) and 900 mm in Smolensk 3 (3rd generation) and Ignalina 2 (2nd generation).

System description

The safety injection system (SI) comprises two subsystems with three independent trains of a 50% capacity each. The safety injection is designed also for long term heat removal.

The SI can provide water to all distribution group headers (3 lines to each header). Each line discharges to the DGH through a check valve located in close proximity to the DGH.

Another set of check valves exists in 2nd and 3rd generations or will be added in the line connecting all the DGH to prevent the ECCS flow injected to one DGH from passing to the other headers. (This flow would otherwise be lost through the 'broken' header, increasing the severity of the event.)

A more detailed description of the SI for the three reference plants (Kursk 1, Smolensk 3, Ignalina and Leningrad 1 and 2) is provided in the following paragraphs.

The planned SI consist of two subsystems, each of these divided in three different trains. The first subsystem (UCS) provides water to the undamaged half of the primary circuit and the second (DCS) to the faulty half of the same circuit.

Both subsystems take suction from the clean condensate tank (3000 m^3 capacity). The UCS consists of three pumps (250 m^3/h maximum capacity) and three independent set of lines to supply water to either the left or the right side of the core. The alignment of the three trains of the UCS subsystem to one or the other reactor side is made through a set of parallel valves which are closed or opened by the initiating signals of the accident.

The DCS consists of three pumps, different from three of UCS which are located in three different lines, which can also provide water to either half of the reactor via the opening and closing of parallel valves performed by the accident initiating signals.

The power supply to each UCS and DCS pump will be provided by a new diesel generator (three will be added of 6.2 MW capacity).

The valves described are electric motor operated valves which can be fed by the corresponding emergency diesel generator.

The water supplied to the damaged circuit will be discharged through the break to the floor of the compartment where it is located. From there it will be pumped to the floor drain treatment system where it can be cooled and cleaned through an evaporator/condenser and then pumped back to the clean condensate tank. The capacity of this system is 40 t/h.

The steam generated in the undamaged part can be discharged to:

- main plant condenser;
- special condenser (fast);
- atmosphere through safety or relief valves.

The description of the steam water discharge was provided verbally and is not included in the given report.

The condensed water could be returned to the deaerator or to the clean condensate tank using for that purpose the condensate pumps.

In Kursk NPP three emergency feedwater pumps (EFP) are now installed as the existing safety injection. Two more EFP will be implemented as an interim measure before the new UCS and DCS are erected. These five EFP will remain as an additional way to supply water to the core.

No credit is given to the main feedwater pumps to mitigate the accident, although these could keep running for about 45 s driven by the turbine rundown. The DBA signal causes the feed valves to close (not shown). The cooling of the undamaged core during the short term is provided by natural circulation (proven to work in some events).

In summary the improvements made to the Kursk NPP (SI) are the following:

(a) Addition of six new safety injection pumps and separation of the mechanical circuits which provide water to the UCS and DCS;
(b) Total separation of each 50% train (except that there is a common source of water);
(c) Addition of check valves to the lines interconnecting the DGHs to prevent the loss of ECCS flow through a broken DGH;

(d) Accumulators;
(e) Increase in the number of EFP from 3 to 5 as an interim measure.

Construction of a water tank under the reactor to collect water in LOCA conditions to avoid reactor system flooding is planned.

Smolensk 3

The main differences of the Smolensk 3 SI from the Kursk 1 are the following:

(a) There are two main sources of water to the ECCS pumps:

- clean condensate tank (3000 m^3);
- accident localization system pools (suppression pool) ($\approx$ 3000 m^3).

(b) The UCS pumps (3) take suction from the condensate cleanup tank, and there are 6 DCS pumps which take water from the suppression pool. These pumps are associated in three pairs constituting for each pair one channel of the SI. It is understood that the capacity of each pair is 50% of that required after an accident. Each pair of DCS pumps is fed by one out of three diesel generators. Each UCS pumps is also fed by one diesel generator.

(c) The suppression pool water cooling system takes suction from the lower pool, takes it to a heat exchanger (cooled by the service water system) and returns it to the upper pool through a system of sprinklers. There are six different loops. Each pair of pumps is fed from a different diesel generator.

The six service water pumps are also fed by these diesel generators.

(d) There are three emergency feedwater pumps fed by three diesel generators (listed in table attached to the document "Use of NPP turbogenerator rundown", and not shown in the mechanical figures of Smolensk 3). These pumps constitute another way to provide cooling to the core from the deaerator.

(e) The steam produced in the pressure tubes will be discharged to:

- Accident localization system (ALS), or
- Main condenser (if in operation).

The pipe discharging steam to ALS is not shown in the available diagrams.

(f) The suppression pool cooling system starts automatically on high pressure in the pool compartment coincident with low steam separator level or pressure difference between pressure headers and steam separator. The same will happen with the service water system.

(g) The water discharged through the break to rooms which are not part of the ALS is taken by the floor drain system and treated and cooled in an evaporator and condenser and then pumped back to the deaerator or to the suppression pool. This system is not shown in the drawings provided, but this information was provided verbally.

(h) The correct functioning of ECCS in case of the DGH break depends upon the correct operation of about 30 check valves, which are not testable during operation.

Ignalina 1

The SI of this plant is quite like that of Smolensk 3, except for the following:

(a) There are no UCS pumps; however, six emergency feedwater pumps are available each supplied by a different diesel generator.

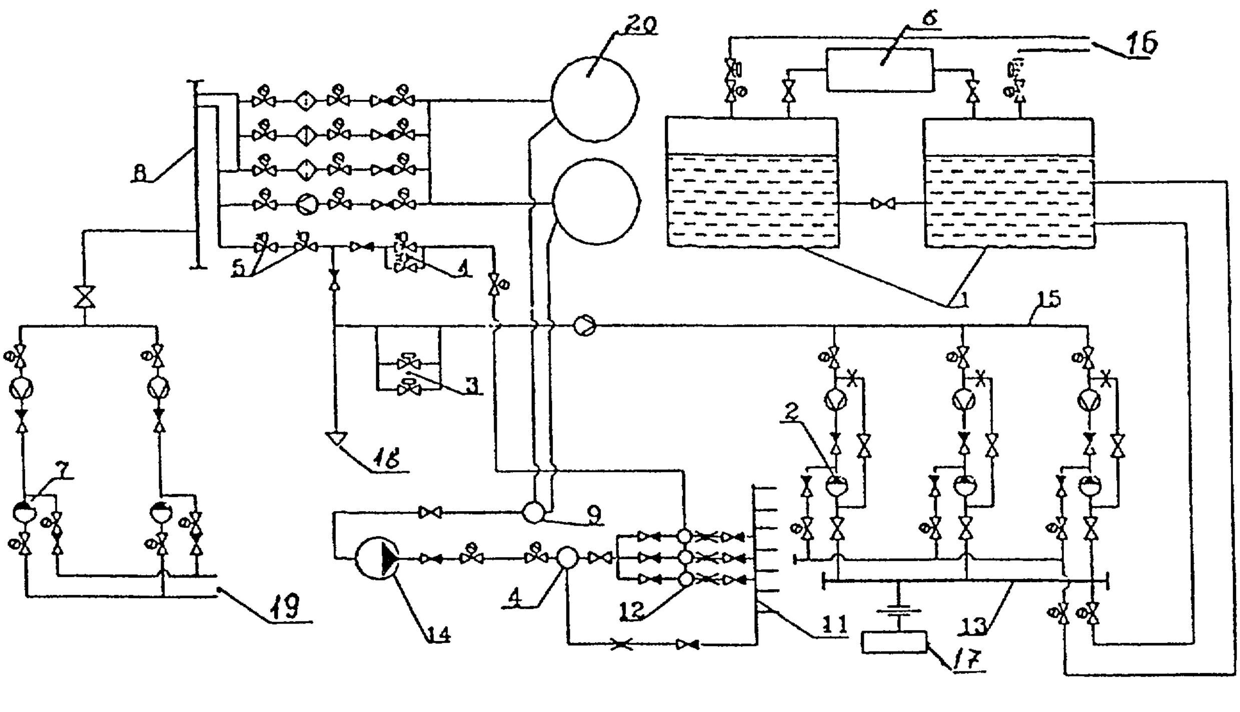
(b) 7000 m^3 of water can be supplied from the chemical treatment plant (desalination system) to the clean condensate tank.

(c) The water needed to cool the damaged side of the core is 750 t/h in the first period after the accident. In addition, 150 t/h are needed to cool the undamaged side.

Leningrad

Referring to Fig. 16 (related to Leningrad 1 and 2):

(a) Emergency cooling systems valves (4) are normally closed; they open to provide flow into the broken loop through ECCS headers and distributing group headers. There are three interconnected ECCS headers in each primary loop that provide water to each distributing group header of that loop.



1. Emergency water tank
2. Emergency feed water pumps
3. Throttling unit
4. ECCS
5. Emergency feedwater supply system
6. Water chemical treatment facility
7. Low-capacity electric feed pumps
8. Electric feed pump pressure header
9. MCP suction header
10. MCP pressure header
11. Distribution group header
12. ECCS header
13. EFP suction header
14. MCP
15. EFP pressure header
16. From main condensate system
17. Core flooding with sea water
18. To second reactor part
19. From deaerator
20. Steam separator

FIG. 16. Emergency water supply system.

(b) The emergency cooling system (3 pumps) has the unique function to inject water in the broken loop; the residual heat from the core accumulates in the confinement system and can be removed by non-safety-grade systems (drainage to chemical treatment facility, not connected to diesel generators).

(c) In case of DBA (break of 300 mm equivalent diameter) 2 pumps are needed to cope with the accident. If an additional single failure is assumed in one distributing group header check valve, all three pumps are needed.

(d) The equipment needed to refill the emergency feed pump tank (from chemical treatment facility or liquid waste storage system, etc.) is safety grade and its active components are served by reliable auxiliary systems (e.g. emergency diesel generators).

(e) The pipe that provides injection to the ECCS header is single and is normally pressurized (limited flow passes through ECCS headers in normal conditions). A break in this pipe either causes primary fluid loss or prevents ECCS injection. During an accident, manual actions are needed in the very short term to line up the system to cool the fuel.

(f) The time elapsed from DBA occurrence and diesel generator startup and alignment on ECCS pumps motors is designed to avoid exceeding the fuel limiting condition (accumulators are not required for DBA break of 300 mm diameter).

(g) During DBA the emergency feedwater supply system (2 pumps) takes suction from deaerators (balance of plant) and injects water in two steam drums. The water circulates in the unaffected half of the core (intact loop) owing to the driving force of the system pumps and to natural circulation (the main circulating pumps trip when the low capacity electric pumps are started). If the main condenser (and/or the turbine) is available, the steam produced is condensed and the loop is closed (normal automatic control systems regulate separators level). If the main condenser is not available, the steam is condensed in an "emergency condenser" cooled by auxiliary systems (safety grade, connectable to the emergency diesel generators). In this case the deaerators have to be refilled in the long term by the chemical treatment facility.

(h) The emergency feedwater supply system valves (5) in the quoted figure are actuated when the water has to be injected in the ECCS header.

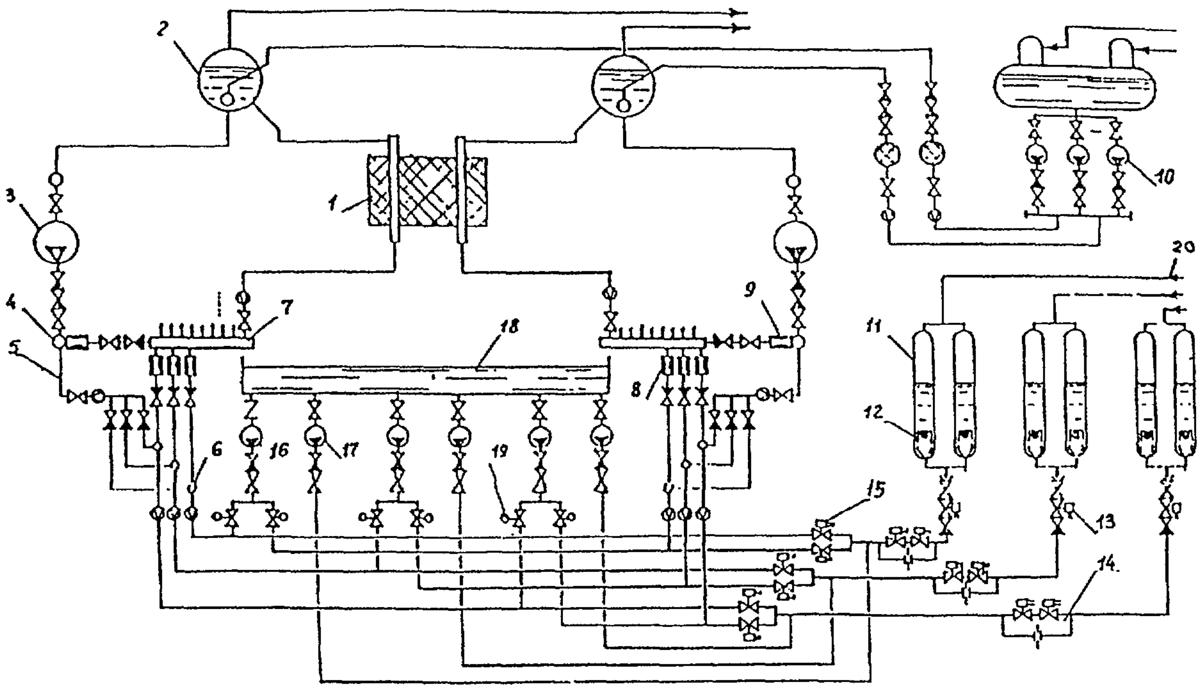
(i) One low capacity electric feed pump is needed to cope with the accident in the case of DBA.

ECCS accumulators and fast acting valves

The ECCS accumulators and fast acting valves act as a high pressure emergency cooling supply for the damaged side of the reactor. They supply cooling to the damaged side of the reactor until the ECCS pumps are operational and able to supply cooling.

The fast acting valves (Figs 17 and 18) are electric motor operated gate valves. These valves are powered by a 'reliable power source' whose ultimate backup is battery power. These valves go from full close to full open in 10 seconds and are open sufficiently to allow injection of cooling water into the core within ~ 3 seconds.

These valves will normally operate with approximately 25 atm of pressure differential across them. They are tested and timed annually with this pressure differential. Note: it was pointed out by an RBMK specialist that these valves have also successfully operated with a much higher pressure differential.



1. Reactor
2. Steam separator
3. MCP
4. MCP pressure header
5. ECCS bypass
6. ECCS header
7. DGH
8. ECCS flow limiter
9. DGH limiting insert
10. Electric feed pumps
11. ECCS storage unit
12. Intercept float valve
13. Intercept fast acting valve
14. Section of intermediate throttling
15. ECCS fast acting valve
16. Pumps for undamaged core side cooling (UCSCP)
17. Pumps for damaged core side cooling (DCSCP)
18. Clean condensate tank
19. Long term cooling subsystem valve
20. Gas supply

FIG. 17. Upgraded ECCS of RBMK 1000 (1st stage of construction).

There are three of these fast acting valves for each channel or division. One is located on the collection header downstream of the accumulators supplying that division; this valve is normally open and is designed to close when the accumulators are drained of water. The closure of this valve is based on the level in the accumulator (for Smolensk 3 it is based on level or time).

The other two valves are located downstream of flow control valves and are normally closed. One valve supplies water to one side of the reactor and the other valve supplies water to the opposite side of the reactor. During an accident the ECCS actuation logic determines which side of the reactor is 'damaged', based on the pressure in the leaktight rooms, and opens only the fast acting valves which supply water to the 'damaged' side of the reactor; the fast acting valves which supply water to the undamaged side of the reactor will remain closed.

The undamaged side of the reactor is cooled by the normal cooling system (main circulation circuit) until the ECCS pumps start supplying flow. The flow is maintained by the momentum (inertia or rundown) of the main coolant pumps.

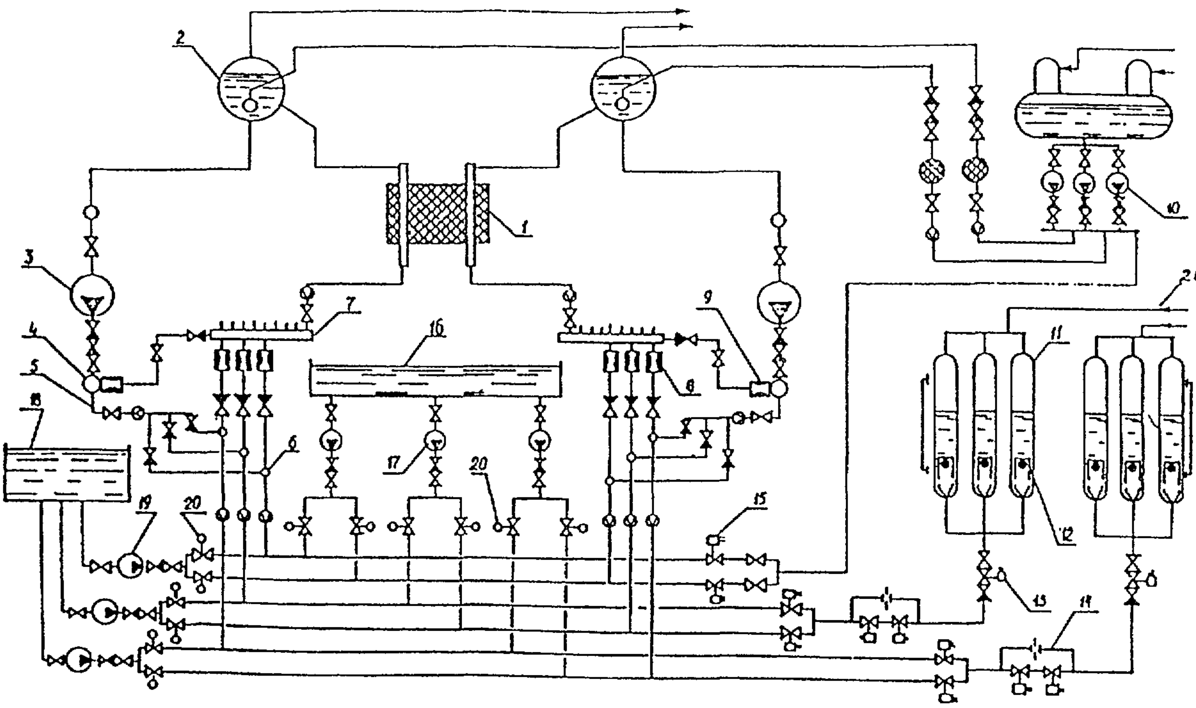
The flow control valves (Figs 17 and 18) for each division consists of two normally open valves in series and restricted flow bypass line around these two valves. These valves are designed to close after about 40 seconds (Ignalina). This allows full flow out of the accumulators for the first ~ 40 seconds and then reduces the flow. This extends the time that the accumulators can supply

water to the reactor to about a total of 2 minutes. The reason the time is extended is to assure sufficient time for the emergency diesel generators to start and then for the ECCS pumps to start and come up to speed and pressure. The power supply for these valves is the same reliable power supply that powers the fast acting valves.

The accumulators (Figs 17 and 18) are basically a high pressure storage tank containing water which is then pressurized by a cover gas (nitrogen). Since there is no physical barrier between the water and cover gas, there is a 'float' valve located at the outlet of the accumulator which should close when the water is drained and thus prevent the cover gas from entering the ECCS piping.

It was stated that experimental tests have shown that this float valve is not affected by the flow of vortices generated in the accumulator when the fast acting valve opens. It was pointed out that there had been a problem with this valve 'sinking' - but a level indication has been added and the level of the float valves is monitored.

It was also stated that there have been no problems in these tests with the closure of the float valve. However, it was pointed out that after it closes it sometimes took considerable effort to reopen it.



1. Reactor
2. Steam separator
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4. MCP pressure header
5. ECCS bypass
6. ECCS header
7. DGH
8. ECCS limiting insert
9. DGH limiting insert
10. Electric feed pumps
11. ECCS storage unit
12. Intercept float valve
13. Intercept fast acting valve
14. Section of intermediate throttling
15. ECCS fast acting valve
16. ALS condensing facility tank
17. Pumps for damaged core side cooling (DCSCP)
18. Clean condensate tank
19. Pumps for undamaged core side cooling (UCSCP)
20. Long term cooling subsystem valve
21. Gas supply

FIG. 18. Upgraded ECCS of RBMK 1000 (2nd stage of construction).

The capacity of the accumulators varies from unit to unit. At Ignalina the total capacity of the accumulators is 200 m^3. At Smolensk 3 and the 2nd generation units the total capacity is 150 m^3. The 1st generation units that have accumulators currently have a capacity of 40 m^3 and all 1st generation units are being upgraded with new accumulators with a total capacity of 225 m^3.

It should be noted that for some units, only 2 of the 3 ECCS divisions have the accumulators and fast acting valves. For these units the other division utilizes flow directly from the feedwater pump discharge header (2nd generation units and Ignalina). The modified or upgraded ECCS for the 1st generation units (at least at Kursk) have the fast acting valves and accumulators for all three divisions.

During our discussions it was noted that there is no reservoir for the nitrogen cover gas; each accumulator is instrumented for pressure.

Testing

Provisions for tests are not shown on the report diagrams; however, it has been stated that during maintenance (at least twice a year) all active components, instruments and their logic channels are tested. A system functional test is also performed at these outages.

Physical hazards to main SI components

Insufficient information is provided on physical hazards to main SI components. It has been generally stated that there is no risk of flooding of ECCS electric components since the drainage system has been adequately designed to cope with any pipe failure in each compartment.

It was also stated that all ECCS electric and I&C equipment is environmentally qualified to perform its function in a harsh environment after an accident.

Protection against the dynamic effects from hazards associated with the breaks of high energy pipelines has not been defined in detail when considering ECCS design. It was, however, generally stated that such a protection exists, and it was agreed to check this item on a plant by plant basis.

III.1.2. Findings

1. The proposed addition of check valves adjacent to the distribution group headers (DGH) for plants of the 1st generation was made on the basis of limiting the discharge of water through the broken DGH coming from the remaining DGH. In this sense the improvement is found acceptable.

2. It is recognized that these valves are necessary in order for the ECCS to successfully cool all fuel channels during some accident scenario. However, it still must be evaluated whether under different (and possibly more likely) scenarios, failure of one or more of these valves could prevent adequate cooling to many fuel channels.

3. The drain system of 1st generation will be upgraded to have additional capacity to cope with flooding in the rooms below the reactor. A new tank will be added to the drain system. To increase this capacity is considered good; however, the basis for limiting this improvement to only this room needs to be examined. The flooding analysis of all rooms which contain high or moderate energy pipelines could be a basic input to determine all the areas where some modification in the drain system could be needed.

4. Prevention of flooding in rooms with redundant equipment (ECCS pumps, etc.) could be an important measure for the operation of the ECCS system.

5. Some supporting systems to ECCS (1st generation RBMKs before reconstruction) have been found necessary. These systems were not designed as safety related systems (regarding power supply, material choice, manufacturing quality, degree of redundancy, etc.). They are, for example, the recirculation of condensed steam to the condensate tank, the drain floor system and recirculation to clean condensate tank (including evaporator, pumps, etc.) or to the ALS pools, the provision of water from the desalination plant, etc. These systems differ from plant to plant.

6. ECCS upgrades - addition of ECCS pumps and piping for the 1st generation RBMKs will provide some redundancy and increase the capacity of the ECCS. However, there was insufficient information relative to whether these upgrades will increase the capacity of the ECCS sufficiently to handle the full spectrum of accidents including a break of the largest diameter pipe.

7. Among the planned improvement measures (Table I), there is the installation of additional water tanks in Kursk 1 NPP. This modification is intended to increase the volume of water available for ECCS injection in the long term.

8. Addition of the number and capacity of the ECCS accumulators will improve the redundancy and capacity of this early injection portion of the ECCS. However, there was insufficient information on whether this increase in capacity will provide sufficient cooling capacity to handle the full spectrum of accidents, including a break of the largest diameter pipe.

III.1.3. Plant specific status

The plant specific status is discussed in Section III.1.1.

III.1.4. Recommendations

1. It should be investigated from the test and performance history of check valves adjacent to DGH how much such incorporation could be an initiator of abnormal events in the primary circuit owing to their failure.

2. Should the result of this investigation be positive (i.e. no significant increase in circuit failure rate is expected), then it is strongly recommended to instal these valves as a first measure on plants of the 1st generation.

3. Testability of these valves should be improved, designing them to be testable during normal operation.

4. An acceptable leak rate though the valves should be defined taking into account the hypotheses made in the accident analyses. The associated actions, should this value be exceeded, need to be specified (decrease power, plant shutdown in a certain time, etc.).

5. A more detailed analysis should be provided, including probabilities of the effect for various accident scenarios both with and without the check valves.

6. The addition of a new drain water collecting tank is found appropriate as a partial measure.

7. A flooding hazard analysis of rooms with safety equipment should be performed, to determine the expected flooding level and to ensure that the safety equipment operation is not threatened by it.

8. Adequate physical separation of safety equipment is necessary to prevent common mode failure due to flooding if needed.

9. RBMK specialists should make additional analyses of the drainage system and other systems used for long term cooling in order to assess their impact on accident sequences.

10. If the drainage system is required to operate after an accident, it should be upgraded as a safety related system (emergency power supply to pump, quality of material and design, tests of equipment, etc.).

11. The need for further improvement in the drainage system should be determined.

12. Perform plant specific analyses to determine the systems and components needed to support the ECCS operation after a design basis accident.

13. Specify upgrading measures for these systems or justify that the core could be cooled without using them.

14. Implement the measures with a priority correlated with their safety importance.

15. An independent review of the outcome of these analyses is recommended.

16. Proceed with the planned 1st phase (interim) upgrade for the 1st generation RBMKs of increasing the number of EFP from 3 to 5, and ECCS lines from 1 to 2.

17. Proceed as soon as possible with the 2nd phase of upgrading the ECCS for the 1st generation RBMKs by installing the additional ECCS pumps (3 for damaged core side cooling - DCS, and 3 for undamaged core side cooling -UCS) and associated diesel generators.

18. A more detailed analysis should be provided showing the specific deficiencies of the original design and how the upgrades remedy those deficiencies.

19. The planned installation of additional water tanks in 1st generation units can improve the capabilities of the plant to cope with large break failures. Therefore, it is recommended to proceed with the above modification.

20. The planned upgrade to increase the number and capacity of ECCS accumulators at the 1st generation RBMKs should be proceeded with.

21. A more detailed specific analysis related to ECCS accumulators should be provided, showing the specific deficiencies of the original design and how the upgrade will mitigate the deficiencies.

III.2. ECCS STARTUP LOGIC

Scope of review: Correct understanding of the logic of ECCS operation during some main accidents.

Document/section reviewed: [10]

III.2.1. Summary of discussions

Pipe ruptures in the most reinforced reactor buildings compartments (leaktight compartments of the second generation reactor units) cause "liquid" leaks that more rapidly discharge the primary water inventory and less effectively reduce the primary system pressure (the specific behaviour depends on the break size). These breaks have a high potential to cause fuel cladding and/or channel tube failures.

(A) If the break takes place in the downcomer part of water lines coming from the steam separator (SD), up to the first check valve, the SD of the affected loop will experience a level drop.

Under the conditions of the downcomer rupture in the reinforced leaktight compartment the ECCS will actuate as follows: the signal of pressure increase is generated initially in the affected compartment (set point 200 kgf/cm^2) which is a symptom of the pipeline rupture (a symptom of leakage) and a signal for SS-5 actuation, and then in about 15 s the signal of level decrease in drum separator (-1000 mm) of the failed half up to emergency setpoint is generated. This signal is the basis of identification of damaged circuit half by logic "AND" operation circuit.

(B) Under the conditions of the pipeline rupture downstream of the MCP check valve, the ECCS system will actuate when the signal of pressure increase in the affected compartment (which is also the signal of SS-5 actuation) and the signal of pressure difference decrease between the MCP pressure header and steam separator drum will coincide in time, since under such a rupture this signal will be generated considerably earlier than the signal of level decrease in the steam separators.

The logic described will actuate the emergency cooling by opening the fast acting isolation valves of all channels of the early injection portion of the ECCS (accumulators; all RBMK units except Leningrad 1 and 2) to feed water to the damaged half of the reactor only.

At Leningrad 1 and 2 (1st generation RBMKs) opening of the fast acting isolation valves and the startup of the emergency feed pumps or low capacity feed pumps (electric) is accompanied by closing of valves to prevent water from being fed to the steam separators of the damaged half of the reactor.

The early injection subsystem of the ECCS is designed to provide the emergency cooling for the first two minutes after the accident. Following that, the subsystem of long term cooling of both damaged and undamaged halves of the reactor starts up.

(C) Pipe ruptures in steam separators rooms or ruptures of feedwater lines coming from steam separator up to the first check valve can cause steam leakages (steam lines), water leakages (downcomers and feedwater lines) or mixed water and steam leakages (risers). For the downcomers and feedwater line breaks, the same logic of paragraph A applies.

In case of steam line breaks, no automatic emergency cooling actuation is envisaged if the leak is inside the control capabilities of normal plant (level/pressure) control systems (a fast scram takes place as a consequence of high confinement room pressure).

If the leak exceeds the control capabilities of control systems, an automatic steam line isolation signal takes place and/or a turbine isolation control valve closure signal takes place, (at a pressure setpoint value of 59 kgf/cm^2 in one SD one turbine group is isolated; at a setpoint value of 54 kgf/cm^2 in the SD, the remaining turbine group is also isolated) possibly followed by an automatic power reduction or SS-5 (shutdown after a trip of both turbine generators of the unit).

For small to medium breaks upstream of the turbine control/isolation valves (energy loss from the break less than the energy produced by the affected half core), after the steam lines are isolated the pressure of the loop stays high, the safety valves may cycle and the level in the drum separator

drops to scram/emergency cooling actuation setpoint of case A above. (The actual behaviour depends on the control/isolation logic). A similar behaviour is envisaged for steam breaks downstream of the turbine control/isolation valves.

For large steam breaks upstream of the turbine control/isolation valves (energy loss from the break higher than the energy produced by the affected half core), the pressure decrease in the drum separator (down to 42 kgf/cm^2) or the high decrease rate (signal implemented in most plants) in an "and" logic with MCPs trip, associated with low flow (5000 m^3/h) through any pump, actuates emergency cooling in both primary loops.

The actuation of ECCS on both primary loops is due to the fact that such large ruptures have certainly seriously unbalanced the steam systems of both turbogenerator groups. In both cases above the steam is freely discharged to the affected building area.

(D) When the reactor is in shutdown condition, with the steam dumped to the main condenser and condenser water returned to the reactor, the pressure in the drum separator has to be maintained in the range 37-74 atm. If the pressure exceeds maximum limit, a signal is generated to open main steam line safety relief valves. If one or two of these valves fail to close after opening, the ECCS will start on the coincidence of a pressure rise signal in the dump line downstream of the valves with either of the following two signals:

- pressure in the SD lower than 42 kgf/cm^2;
- flow rate decrease in any main coolant pumps down to 5000 m^3/h, given a previous pumps trip.

The above described logics A, B, C unless differently specified will actuate the emergency cooling in the broken loop and disable the actuation in the unaffected loop.

The actuation of the intact loop cooling will take place only if normal feedwater system is no longer able to carry out its normal operation or if the main turbine/condenser will no longer be available.

The signals to actuate emergency cooling (low capacity feed pumps or undamaged circuit pumps depending on the plant generation) are normal feedwater pumps trips in coincidence of low SD level (200 mm below the separator axis in a 1/4 logic) and low pressure in the feedwater pressure header (76 bar in a 2 out of 3 logic).

III.2.2. Findings

1. A comprehensive understanding of the emergency cooling systems actuation logic depends strongly on the availability of detailed logic diagrams for each plant generation and of:

   - main coolant pumps trip logic;
   - isolation system trip logic;
   - main cycle control system trip logic.

2. In Table I improvements on signals which are part of the scram logic are included. In Ref. [12] a recommendation for beyond DBA is included that addresses the use of independent group sensors for actuation of different safety functions (scram and emergency cooling). No diversification of physical parameters seems to be envisaged for initiation signals of emergency core cooling systems, often in "and" logic.

III.2.3. Plant specific status

The logic generally applies to plants of all three generations. Differences could be found in set-point values and in the details of implementation.

III.2.4. Recommendation

1. In going towards a signal modification in protection system logic, it is recommended to plan a wider programme of systematic review to implement other effective changes, including:

   - diversification of signal for ECCS startup;
   - improvement of the diversification of sensors that actuate different safety functions, also from the point of view of beyond DBA protection.

III.3. SEPARATION

Scope of review:	Understand the functions of the ECCS and confinement system.
Document/section reviewed:	[10], [11], [12], [13]

III.3.1. Summary of discussions

Following is a very brief description of the degree to which the various channels or divisions of ECCS and confinement system components are isolated or separated from the components in other divisions. This item has not been discussed very much and was not mentioned in the supporting documents that were provided, so there is very little detail.

There is at least some recognition for the need to separate the various ECCS and confinement system divisions. For example, the piping for the various ECCS divisions contains check valves when lines from different divisions are connected to common headers (the distributions group header for example). The three ECCS divisions each have their own separate piping systems with the only common points being the water supply (the water storage tanks are indicated to be interconnected) and the connection to the main circulation circuit.

It is not clear whether this piping is physically separated (i.e. in separate rooms). It is also not clear whether the pumps, accumulators and fast acting valves are physically separated from those components of the other divisions. It was stated that in some parts of the plant there is separation, but not in others. There was no specific design intent concerning separation.

The emergency diesel generators are located in separate rooms; however, there was no intent to separate the electrical cables. The exception is Smolensk 3 and to a lesser degree the Ignalina units where separation was considered to some extent during the design.

Note: The electrical separation issue is also addressed in Appendix IV.

III.3.2. Finding

1. The RBMK design has not provided for full physical separation of the ECCS components and equipment for one division or channel from the equipment and components of another division. There was insufficient information provided to ensure that components from multiple divisions would not be adversely affected by a common event (fire, flood, falling equipment or debris, etc.).

III.3.3. Plant specific status

As indicated above.

III.3.4. Recommendations

1. More detailed analysis of how the equipment is adequately protected from common events (common mode failure) should be provided.

2. If item 1 above cannot be done, specific fixes that will provide adequate protection against common mode failures (i.e. improved fire protection, shielding, etc.) should be identified.

3. Plant specific missions should be carried out to investigate this topic.

III.4. ECCS PERFORMANCE CRITERIA

Document/section reviewed: [10]

III.4.1. Summary of discussions

The following criteria were presented as those which indicate acceptable ECCS performance during all design basis accidents:

- the fuel element cladding temperature should not exceed 1200°C;
- the local depth of fuel element cladding oxidation should not exceed 18% of the original wall thickness; and
- the amount of oxidized zirconium in the core should not exceed 1% of the original mass of zirconium in the fuel sheaths in the core.

III.4.2. Findings

1. The use of the 1200°C maximum cladding temperature criterion for a channel type reactor such as the RBMK may be overly restrictive. The criterion was derived by the USNRC for pressurized and boiling water reactor designs.

2. In the RBMK design two 18 element, 3.5 m long fuel bundles are located inside each Zr 2.5 Nb pressure tube (80 mm in diameter with a 4 mm wall thickness) which in turn is surrounded by graphite block (250 mm x 250 mm). Under normal operating conditions the maximum graphite temperature remains less than 700°C. The presence of large quantities of graphite (1700 tonnes) provides a potential heat sink for overheated fuel and a physical barrier between the fuel bundles.

III.4.3. Plant specific status

Not applicable.

III.4.4. Recommendation

1. Safety experts for channel reactors, PWRs and BWRs should meet to review whether the use of PWR/BWR ECCS performance criteria in the safety assessment of a channel type reactor such as an RBMK is technically appropriate.

III.5. SELECTION OF LOSS OF COOLANT DESIGN BASIS ACCIDENTS

Document/section reviewed: [11]

III.5.1. Summary of discussions

All pipe breaks, analysed as design basis accidents, whether fuel channel, distribution group header or pressure header, are considered to be guillotine failures with double ended discharge. Partial pipe breaks are considered to be highly improbable in that any partial break capable of heat removal deterioration has a size exceeding the critical, so it is assumed to result in guillotine double ended pipe failure.

The probabilities of the various sizes of pipe breaks are stated to range from $1.2x10^{-6}$ occurrences per reactor year for fuel channels to approximately 10^{-8} occurrences per reactor year for the rupture of a pressure header with failure of the check valve on one distribution group header.

There are many valves present in the heat transport system. As an example there is a control/isolation valve associated with each fuel channel, a check valve and a stop gate valve associated with each distribution group header, and check and throttling and stop gate valves in the piping associated with each main circulating pump.

III.5.2. Findings

1. Partial breaks, of a size that may or may not be stable, could result in periods of flow stagnation in one or more fuel channels. These periods of channel flow stagnation could result in cladding temperatures in excess of those quoted in the referenced DBA analysis for RBMKs.

2. The probabilities of pipe failures cited in the RBMK reports are considerably lower than those presently in use in the safety analysis of other reactor designs. Also failure of one or more valves in the heat transport system could result in an additional range of break discharges from the heat transport system.

III.5.3. Plant specific status

Applicable to all RBMK units.

III.5.4. Recommendations

1. Fuel and fuel channel transient thermal/mechanical behaviour and consequent fission product release should be assessed over the entire range of break sizes for all the various pipes and headers in the heat transport system. Although potential breaks of a significant size may not be stable, the assessment of such partial breaks, and designing safety systems to adequately cope with the predicted consequences of the entire break spectrum (including partial breaks), is necessary according to the internationally accepted "defence in depth" philosophy.

2. The technical basis for the pipe break probabilities cited in the RBMK experts' reports should be rationalized with the conservative failure probabilities assumed in the safety assessment of other reactor designs.

3. The potential for valve failures and/or failure of the welds/connections associated with valve installation should be assessed. The range of failure modes and failure probabilities for the valves within the heat transport system should be studied in detail.

4. Channel reactor, PWR and BWR technical safety specialists should have further discussions with the RBMK safety specialists to rationalize the definition of loss of coolant design basis accidents for RBMKs.

III.6. LOSS OF COOLANT ACCIDENT ANALYSIS

Scope of review:	ECCS; Containment/confinement system.
Document/section reviewed:	[10], [11], [12], [13]

III.6.1. Summary of discussions

The analysis of the loss of coolant accident (LOCA) is traditionally one of the accident analyses used to confirm the effectiveness of the shutdown, emergency core cooling and containment/confinement systems. This part focuses on the latter two systems only, as the shutdown system effectiveness is addressed in Appendix I.

In general, one should acknowledge the various recommended design improvements to the safety systems. These include the improvements to the emergency core cooling system (ECCS), to the confinement system (eg. the steam dump pipes to protect the reactor cavity from overpressure) and to the reduction of positive void reactivity. With respect to LOCA analysis, computer codes with an apparently sound technical basis are used to perform the analyses. Furthermore, a beyond design basis LOCA is also analysed for the purpose of accident management planning. The defence in depth approach is also evident, with the primary emphasis on prevention, followed by mitigation and the protection of the multiple physical barriers against the release of radioactive materials.

III.6.2. Findings

There are a number of subject areas needing further technical information. This is not unexpected owing to the limited review time and the summary like nature of the documents provided. It should be cautioned that definitive conclusions should not be drawn without having reviewed the clarifying technical information.

1. The design basis LOCA considered in Ref. [2] to be the worst is the rupture of the pressure header with the failure to close of check valve of one distribution group header (DGH), although it is acknowledged in the same reference that partial pressure header rupture could yield worse consequences, possibly owing to potential core channel flow stagnation. However, one could envisage other LOCA accidents such as steamline failure posing other unique problems, because in the absence of a main steam isolation valve, there could be irrecoverable loss of coolant inventory and unmonitored release of radioactive materials. A pipe/header rupture together with a seismic event could also pose its special problems if some of the mitigating systems are not adequately seismically qualified. Therefore, in order to ensure that the worst design basis LOCA is identified for a particular nuclear power plant, a systematic review of the plant should be carried out, including but not limited to a survey of the critical pipe/header break size and location.

2. It should be noticed that RBMK specialists believe that partial pressure header ruptures are highly improbable since crack sizes are greater than the critical crack size.

3. To demonstrate that ECCS equipment and confinement components will not be disabled by cross-link effects in the short as well as in the long term still needs to be justified. Cross-link effects include environmental effects such as flooding, condensation, temperature, pressure and radiation, or dynamic effects such as water hammer, pipe whip and fluid jets.

4. The extent to which ECCS and confinement equipment are seismically qualified has not been mentioned in Refs [10-13]. It is particularly important for equipment which is required for long term operation, such as the water and power supplies for long term ECCS operation and the system for confining the radioactive releases.

5. Only limited information was available on the exact configuration of the ECCS and its credited mission time. During the discussion, it appears that long term ECCS could utilize the liquid waste management system, use the leakage collection system as a long term recovery system or draw water from the sea/lake as a once-through system. It is important to identify clearly the essential water and power supplies for long term ECCS operation.

6. The many probability and reliability claims in Refs [10-13] require a review of reliability analysis and applicability of data.

7. Only limited information was available on the following:

   (a) airborne and possibly waterborne leakage of radioactive materials from areas such as confinement compartment envelope, safety relief valves, ECCS equipment, particularly long term ECCS equipment;

   (b) assumptions on radioactive releases, such as fission product distribution, washout, plateout and charcoal filter efficiency;

   (c) assumptions for exposure calculations such as weather scenarios, location and exposure time duration of the most exposed individual.

8. For the computer codes used in the LOCA analysis, assessment of the following is still required:

   (a) the physics, mathematical models, empirical correlations and simplifying assumptions;

   (b) validation, including benchmarking against standard or analytical problems, comparison against pertinent experimental data and actual RBMK transients during commissioning and operation and 'blind' testing against relevant test data.

III.6.3. Plant specific status

Not applicable.

III.6.4. Recommendations

1. An independent review of the studies carried out by RBMK specialists in order to identify the worst design basis LOCA should be organized. (See also Appendix III.1.)

2. Separate reviews in the form of specialist meetings should be held to clarify the following subject areas:

   (a) cross-link damage;
   (b) reliability analysis;
   (c) radioactive release calculation;
   (d) computer code documentation and validation standards;
   (e) human factor engineering and operator models.

   It is further recommended that sufficient documentation should be prepared and made available to the specialists prior to the meetings.

3. The exact long term configuration of the ECCS and its credited mission time should be presented for review.

4. Further technical information should be presented and independently reviewed on the reliability assessment for the gas system used to maintain an inert environment around the graphite in a RBMK reactor.

III.7. ACCIDENT MITIGATION

Scope of review: Beyond design basis accident.

Document/section reviewed: [12]

III.7.1. Summary of discussions

The discussions began with a "reading" of the respective report (Ref. [11]). Major discussions and questions involved the information in Table 3 of Ref. [11]. This table is part of a PSA level 1 study carried out for Leningrad 1. An attempt was made during the meeting to review this report. However, this review was not within the planned scope of the meeting and it was difficult for RBMK specialists to fully respond to some specific questions. The numerical values of Table 3, Ref. [11] were not clarified at the meeting and therefore they could not be used as a technical basis and justification for decisions on RBMK safety.

There was considerable clarification provided relative to the functioning of the ECCS.

In addition, the discussions provided considerably more information on how all the various "Recommendations" are being handled. The following information was provided:

- There is an "overall plan" that is reviewed and updated routinely - for example, the ECCS beyond DBA analysis was a specific line item on the "overall plan". IAEA asked for a copy of this "overall plan".
- The utilities are responsible for evaluating the benefit of the recommendations and whether to implement them or not (there is also some input from the regulatory authorities concerned).
- All modifications/upgrades must be reviewed against the safety analysis. The modification package along with this review must be reviewed and approved by the regulatory body prior to performing any modifications.

III.7.2. Findings

1. In order to cope with beyond design basis accidents, a recommendation has been focused by plant designers to design and install an independent source of water supply to the primary circuit. The purpose is to provide long term cooling in case of failure of all systems.

2. Among the proposed improvement measures identified in Table I, there is a modification in this area applicable to each considered plant.

3. It has been proposed to provide an independent line for water supply from external sources (lake, etc.) in all plants considered.

4. No detailed information has been made available about the requirements applicable to this additional equipments.

5. The technical basis and justification for the numerical values of Table 3, Ref. [11] were not available.

III.7.3. Plant specific status

As far as independent water sources are concerned, the first generation units (Kursk) already have diverse water sources (service water connection and artesian wells); the other generations still do not have these features implemented, (Table I).

III.7.4. Recommendations

1. The possible availability of large amounts of water can improve the long term plant response to specific scenarios in which either active components fail or water sources are exhausted.

2. The planned actions have the potential to considerably increase safety along the lines followed in other countries.

3. The design implementation should take into account:

   - interfaces with other systems and with layout restraints;
   - interfaces with operator teams in the reference design conditions;
   - the need to develop, in parallel, suitable accident management procedures for the above reference conditions.

4. A peer review of the Leningrad 1 PSA level 1 is necessary to justify results given in Table 3, Ref. [11] prior to use of these results in support of safety decisions. The IAEA should undertake this review with the help of consultants if requested by the relevant Russian institutes.

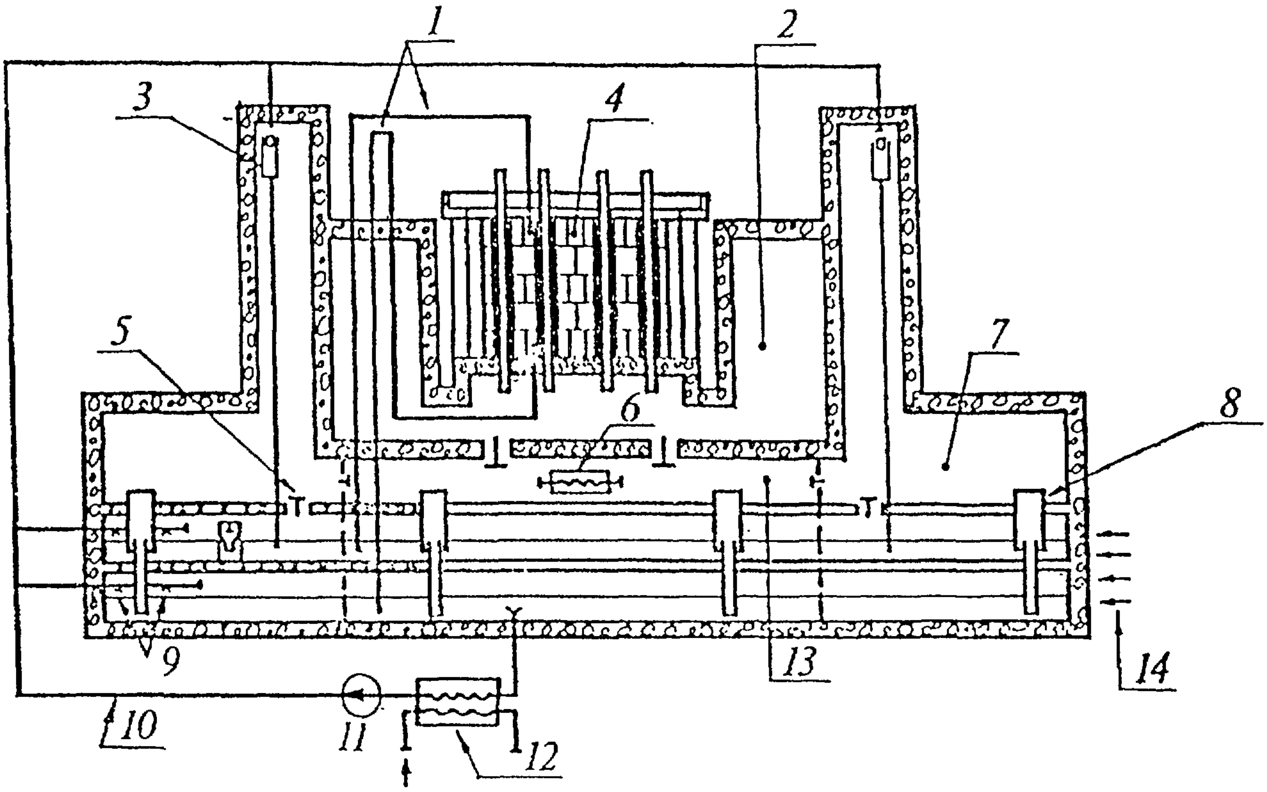
III.8. CONFINEMENT SYSTEM

Scope of review: Confinement system of Smolensk 3.

Document/section reviewed: [13]

III.8.1. Summary of discussions

Only a part of the primary circuit is enclosed in an accident localization system. This system consists of more or less leaktight compartments. All main pipelines, headers and components carrying subcooled water are part of the accident localization system (ALS). There are two major parts of rooms of the ALS: the compartment system of the rooms below and to the side of the reactor cavity in which the inlet pipelines to the technological channels and the groups distribution headers are located (24 100 m^3, 0.1 m^3/h leakage, 0.8 kgf/cm^2 design overpressure) and the compartments system where the main coolant pumps and the suction and pressure headers are located (19 500 m^3, 0.6 m^3/h leakage, 4.4 kgf/cm^2 design overpressure). The tightness is measured once every four years.



1. Reactor cavity protection system against excess pressure
2. Main circulation circuit piping room
3. Air ejector
4. Reactor cavity
5. Relief valve
6. Condenser
7. MCP room
8. Condensation facility
9. Sprinklers
10. Cooling system
11. Pump
12. Heat exchanger
13. Steam distribution corridor
14. Two storey pressure suppression pool

FIG. 19. Schematic diagram of the accident localization system.

The (civil engineering) design overpressure of the pressure compartment system amounts to 4.4 kgf/cm^2 whereas the calculated pressure at the maximum design basis accident (rupture of a header of 900 mm in diameter) is given by 3.5 kgf/cm^2 (absolute).

The reactor cavity (Fig. 19), the main coolant pumps (MCP) and main header rooms, the lower water line rooms and the steam distribution corridor are connected to a pressure suppression pool system by 236 steam dumping channels (total cross-sectional area: 50 m^2) immersed in its water space.

The pressure suppression system consists of two floors of concrete compartments with steel liners. The water level is 1.2 m in each compartment. The total water volume, which is also the ECCS water, equals 3200 m^3 whereas the free space of the pressure suppression pool is 3700 m^3.

In addition to a system of valves between the compartments of the pressure suppression system, the localization equipment includes also a sprinkler system with pumps and heat exchangers and surface condensers in the steam distribution corridor between the MCP compartments.

The rooms where the steam-water lines, the drum separators and upper parts of the downcomers are located are not included in the accident localization system. These rooms are in

connection with the main reactor hall by an unsealed area of about 5 m^2. The argumentation of the designer is that these compartments were not included in the ALS because ruptures of tubes here are of little radiological impact, when no previous fuel cladding failures have occurred. It is assumed that leakages and ruptures of primary pipelines in these rooms can be coped with by the ventilation system and its special filter system. In addition, the ventilation system feeds into the stack system with its own filter system. The motors of this ventilation system are connected to the reliable power supply system. Heat exchangers are supposed to cope with the released energy from the primary system. The steam-air mixture is drained to the purification system. According to the presentations, the design leak in this part of the primary circuit with reference to the compartment system is the rupture of a pipeline of 300 mm in diameter.

To protect the steam separator/reactor hall compartment system from over-pressurization, safety relief flaps are installed which open at an absolute pressure of 1.015 kgf/m^2 ($\approx$ 1.015 bar) to the atmosphere.

III.8.2. Findings

1. The defence in depth concept is not consequently applied to the rooms in which the steam-water part of the primary circuit is located. This holds also for the reactor hall.

2. However, investigations regarding the strengthening of the reactor building structure, including the roof of the reactor hall, are being discussed by the RBMK specialists including the installation of a leaktight compartment system for these rooms (steam drum separator rooms, steam/water pipeline rooms, central reactor hall). Thus an additional barrier for a part of the primary circuit could be established in conformance with international practice as well as with the defence in depth concept.

III.8.3. Plant specific status

No units of the 1st generation of RBMKs have confinement and pressure suppression systems similar to those of plants of the 2nd and 3rd generation, but separate buildings for pressure suppression systems are in discussion for these units.

III.8.4. Recommendations

1. The decision to create a pressure suppression system for 1st generation units which is similar in its design efficiency to those of the units of the 2nd and 3rd generation is supported. Since these devices were not foreseen in the original design, the construction of a separate building seems to be an appropriate way.

2. It is recommended to upgrade the upper rooms (reactor hall, steam separator rooms) with respect to leaktightness and confinement function in order to have an additional barrier in accordance with the defence in depth concept.

3. It is suggested that the discharge from these rooms in the event of a major break in the primary system within these compartments could be directed into the pressure suppression system or to a filtered ventilation.

4. A stress analysis and a seismic analysis should be performed before the reconstruction of these rooms.

5. In accordance with international practice and in application of the defence in depth concept to the confinement function, consideration should be given to the installation of steamline isolation valves (see also Appendix III.4.)

III.9. REACTOR CAVITY OVERPRESSURE PROTECTION SYSTEM

Scope of review:	Brief description of the functions of this system for first and second generation RBMK plant design, plus descriptions of Russian plans for improvements and status.
Document/section reviewed:	[10], [13]

III.9.1. Summary of discussions

The purpose of this system is to avoid pressure increase in the enclosed, sealed core region of the reactor (denoted "reactor cavity" by the RBMK specialists), which might otherwise cause damage to the boundary of the sealed space. In particular, the greatest concern is the lifting of the upper biological shield/head structure as occurred during the Chernobyl accident. According to information in Ref. [13], this places an upper limit of 3.1 atm absolute pressure in the cavity.

The source of pressurization is postulated to be failure of a process channel inside the cavity region. Such a failure would result in expansion of a steam-water mixture into the confines of the cavity. To mitigate pressure buildup, there are a total of eight 300 mm emergency steam discharge pipes, four venting from the space above the graphite stack and four venting from below it. The steam discharge pipes connect to systems designed for steam condensation.

The design basis for the cavity overpressure protection system is the failure of one process channel. The design of the discharge-condensation system is based on conservative assumptions on steam-water blowdown for the single tube failure; that is, a complete guillotine break with double ended flow. Factors such as flow resistance due to the moderator and shield block stacking and the presence of fuel assemblies inside the channel were ignored in the design basis analyses for conservatism.

The overpressure protection system is different in the first and second generation RBMK designs (i.e., without and with pressure suppression pools, respectively). In the first generation plants, the discharged steam is condensed in a heat exchanger, and non-condensable gas enters a holdup tank before being released to the building stack via the gas cleanup system. In the second generation plants, the discharged steam is piped to separate zones of the bubbler-condenser pool where it is condensed. The bottom of the pipes are submerged approx. 2 m in the pool, so there is a 2 m water seal between the cavity and the pool which must be exceeded in order to initiate steam discharge.

Information in Ref. [10] indicates that the actual capacity of the steam discharge-condenser systems actually exceeds the design basis of one channel failure. The analyses were based on not exceeding an ultimate pressure of 3.1 atm in the cavity. On this basis, the actual existing capacity is said to be two tube failures for first generation plants (Leningrad 1 and 2; Kursk 1 and 2; and Chernobyl 1 and 2) and 3 tube ruptures for second generation plants. Smolensk 3 was constructed with improvements already in the design and can accommodate nine simultaneous tube ruptures. The analysis methodology was the same as described for the design basis accident.

Improvements to the reactor cavity overpressure protection system are planned for all units (except Smolensk 3) and have been initiated at some units. These improvements are intended to increase the number of simultaneous tube failures that can be accommodated without exceeding the ultimate pressure limit of 3.1 atm within the cavity. There were no technical requirements given as the basis for the improvements, and the DBA for this system remains a single channel failure. The intent of the improvements is to make the safety margin as large as reasonably achievable.

The first stage of improvement is to add two atmospheric relief valves and pipelines connecting to existing steam discharge pipes at the top region of the reactor. Each relief pipeline is

400 mm in diameter and terminates in a special valve. The valve is said to have a calibrated aluminum rod which ties the valve cap to its seat. The design pressure to fail the calibrated rod is said to be 2.8 atm, 0.3 atm below ultimate allowed pressure. When the rod fails the cap is lifted, and blowdown of the gas/steam mixture to atmosphere ensues. It was stated that the valve cap would return to its seat by its own weight when the pressure is reduced, but that pressure level is not known. There is no requirement that the valve would reseal upon closing. The valve is said to have a very reliable opening pressure, although it is not possible to test or verify as installed.

This first stage of improvement was said to increase the accommodation of simultaneous process tube ruptures by one or two. Specifically it was said that for Smolensk 2 this upgrade will be made during the next shutdown which will increase the number of simultaneous ruptures withstandable to four (from three currently). This improvement is planned for all units. It was said that this upgrade has already been completed at Leningrad 1 and Chernobyl 1 and 3 and is under way at Kursk 2.

The second stage of improvement is to add an additional 600 mm diameter steam discharge pipe in the upper region of the reactor. According to information presented, this new vent capacity will be extended to the pressure suppression pool system for the second generation units. This pipe also is dedicated to the atmospheric vent function. Two 400 mm diameter pipes are connected to the top (T joints) of this new steam discharge pipe so that the same atmospheric relief valves described above will also be used on this new pipeline. The total number of simultaneous process tube ruptures is said to increase to nine for all units when both stages of improvement are taken into account.

Some consideration had been made to the environmental impact from possible direct releases. Briefly it was said that the consequences were acceptably small since only very limited fuel failures were foreseen.

An additional major improvement was mentioned pertaining to the first generation units. There is a plan for an additional stage of improvement involving construction of new pressure suppression pools in new sealed buildings. The atmospheric discharge pipelines would be routed to this sealed space for steam condensation, thereby eliminating the need for the atmospheric vents. This seems to be at the proposal stage and does not appear to be scheduled at any of the units.

In general, the capacity of the overpressure protection system is being re-evaluated taking into account actual experience at Leningrad NPP. The extreme conservative analysis approach taken now in the rating may be relaxed on the basis of this experience. It can be expected that the rating in terms of simultaneous tube failures will be increased in the future on this basis.

III.9.2. Findings

1. The RBMK specialists conclude, on the basis of their DBA analyses, that there are no sequences that result in multiple tube failures inside the reactor cavity. This is a very important conclusion for this type of reactor owing to the serious consequences of overpressurizing this cavity. This conclusion needs to be peer reviewed. The types of accident sequences analysed and the results of those analyses need to be subjected to peer review and audit calculations.

2. Owing to the fact that there is no design basis for multiple tube ruptures, it has not been possible to assess the merits of the improvements described for the cavity overprotection system. These improvements, after implementation of all stages, will provide conservatively based capacity for nine simultaneous tube ruptures.

3. The first stage of improvement, which involves the addition of two atmospheric vent valves connected to existing steam discharge pipes, was said to add a one or two tube failure vent

capacity to the current capacity of two or three tube ruptures. This amounts to a very small incremental gain and raises the question of the net benefit, in view of the fact that it is achieved by opening the cavity space to the atmosphere and the inherent uncertainties in valve opening and closing characteristics.

4. The second stage of improvement, which involves addition of a new 600 mm steam discharge pipe from the cavity to the condenser pool, plus two additional atmospheric dump valves, was said to result in a total capacity of nine tube ruptures.

5. The atmospheric vent valves are said to open very accurately at 2.8 atm absolute pressure compared to the maximum safe pressure of 3.1 atm in the cavity. The consultants have concerns about the functioning of the valve, i.e. what is the uncertainty in the range of opening pressure with the calibrated rod installed in the assembly; can it be reliably assured that it will reclose after venting, will it reseal after opening? This cannot be functionally tested in situ.

6. There is a proposal for an additional major improvement which would obviate the need for atmosphere venting; that is, the construction of separate sealed buildings containing water pools for the steam discharge pipes entering them.

III.9.3. Plant specific status

The plant specific status is given in Table XVI.

TABLE XVI. REACTOR OVERPRESSURE PROTECTION

Unit	Improvements		Current Overpressure
	1st stage	2nd stage	Relief Capacity
Leningrad 1	Completed	Planned	4 tubes
Leningrad 2	Under way	Under way	9[a]
Kursk 1	Planned	Planned	2
Kursk 2	Under way	Planned	4[a]
Chernobyl 1, 3	Completed	Not planned[b]	4
Smolensk 3	Completed	Completed	9
Other units	Planned	Planned	3

[a] Capacity following completion of improvement, unit restart.
[b] Not currently planned for Chernobyl owing to uncertainty of continued operation.

III.9.4. Recommendations

1. The measures being taken to increase the safety margin of the cavity overpressure protection system were recognized. However, the solution of uncontrolled venting of the cavity space to the atmosphere was questioned. The recommendation is to encourage the technical solution of greater capacity by the fast condensation in pressure suppression pools in leaktight areas already proposed, as exists in Smolensk 3.

2. A topical specialists meeting should be organized to review the design basis for the reactor cavity overpressure protection system concerning multiple tube failures, review in detail the

technical solutions, and assess the adequacy of both the design basis, the technical solutions, and the best estimate protection capability measures. To accomplish this:

(a) Recommendations for specific accident sequences to be analysed should be compiled in advance of the meeting. This could focus on those sequences thought to be most likely to challenge pressure tube integrity involving multiple tubes. The meeting should focus on RBMK presentations of results of requested analyses as well as selected audit calculations from the international community. Clearly the interest is to confirm the position of the RBMK specialist that multiple tube ruptures do not occur. This is regarded as high priority and should be scheduled for early action.

(b) The meeting should include the topic of pressure tube integrity, including the response of pipe and pipe welds to the rapid transient conditions of accident sequences.

(c) The RBMK specialists should provide information about the specifications of improvements including requirements, technical solutions, drawings, testing and experience. It should be clear what is implemented and planned at the various units.

(d) The RBMK specialists should provide assessment of best estimate overpressure protection capacity based on experience as well as the capacity based on conservative analysis methodology.

III.10. HYDROGEN CONTROL

III.10.1. Summary of discussions

One of the subsystems of the accident localization system is designed to remove the non-condensable gases including hydrogen. This is shown as item 3, air ejector, on Fig. 19.

III.10.2. Findings

1. This system consists of 3 subsystems, two of which are in operation, and the third is a spare. One subsystem serves the leaktight rooms, and the other serves the under reactor rooms. Each of these subsystems contains two blowers, hydrogen detectors and hydrogen recombiners.

2. The non-condensable gases are drawn out of the various rooms or chambers and directed through charcoal filters before being released out the stack. When the hydrogen concentration exceeds the set-point, an alarm is activated in the control room. An operator must then start the hydrogen recombiner.

III.10.3. Plant specific status

Not applicable.

III.10.4. Recommendation

1. An unacceptable concentration of hydrogen could be present in the confinement zones, especially following severe accidents. For accident mitigation and accident planning, it is recommended that an automatic hydrogen control and mitigation system should be designed and installed.

III.11. CONTAINMENT TESTING

III.11.1. Summary of discussion

The general rules for nuclear plant safety provisions regulating the RBMKs state that for localization systems, commissioning tests with regard to pressure and tightness capability should be carried out at design pressure. The rules also state that periodic tests should be made after the plant has been taken into operation.

It was said by the RBMK specialists that such commissioning tests had been performed on all RBMK reactors with the exception of the two units at Ignalina. There the tests had, for reasons not explained, been performed at substantially lower pressure levels than design.

III.11.2. Finding

1. Commissioning tests made only at low pressure leave uncertainties with regard to the integrity and the leakage rate characteristics in case of pressurization for the Ignalina localization systems.

III.11.3. Plant specific status

Not applicable.

III.11.4. Recommendation

1. The situation with regard to pressure capacity and leaktightness for Ignalina localization systems should be explained and justified.

III.12. MAIN STEAMLINE FAILURES

III.12.1. Summary of discussion

In this accident scenario, a guillotine rupture of a 600 mm diameter steam pipe is postulated, resulting in rapid depressurization of the main reactor cooling circuit. Assumption of a coincidental station blackout further exacerbates the accident scenario. When the cooling circuit pressure drops to 38 bar, circulation stops and emergency core coolant injection is required for both parts of the reactor cooling circuit. The ECCS actuation signals and logic is depicted in Fig. 6 of Ref. [10]. Steam pipe ruptures inside as well as outside tight rooms are assessed. In the first case, reactor scram (SS-5) is initiated by either increased pressure in the leaktight room or loss of electric power, whereas in the second case, reactor scram is initiated by trip or unloading of both turbines.

During such a postulated transient at Smolensk 3, the rate of steam discharge is calculated by the CRITICA code, accounting for critical steam discharge where necessary. Due to the presence of special nozzles, the initial discharge rate at 100% is limited to 1920 kg/s. Since electrical power is assumed lost, all main and feed pumps stop. The main circuit pressure drops rapidly to 43 bar in 40 seconds during which cooling is provided by pump rundown; thereafter, the reactor is cooled by the ECCS and fuel cladding integrity is maintained. Reactor power decrease monotonically following fast shutdown in 2 seconds. In-core voiding occurs in 6 seconds. Steam discharge terminates in approximately 2 hours when the system pressure reaches atmospheric level. About 650 tonnes of

steam are discharged. There is no main steam isolation valve. As a matter of fact, such valves are never considered in the design, partly because the normal operating iodine 131 activity level is deliberately kept low at 3.7×10^3 Bq/L (10^{-7} Ci/L) by prompt detection and removal of defective fuel (the regulatory limit is 3.7×10^5 Bq/L (10^{-5} Ci/L)).

III.12.2. Findings

1. Radioactive releases from ruptured steam pipes are an unmonitored and uncontrolled release of radioactive material and is therefore undesirable. While the normal operating I-131 level is low, the level could increase by as much as ten fold during the accident, resulting perhaps in a not insignificant release of radioactive material.

2. It is difficult to believe that the integrity of the leaktight room can be maintained if the main steam pipe ruptures inside it. The consequence of structural failure of leaktight rooms should be investigated.

III.12.3. Plant specific status

Not applicable.

III.12.4. Recommendation

1. In order to minimize the unmonitored release of radioactive materials to the extent practical, it is recommended that the installation of main steam isolation valves be considered. Due consideration should be given to the safety advantages and disadvantages of such valves.

III.13. SITE BLACKOUT

III.13.1. Summary of discussion

Following a site blackout, control rods drop into core under their own weight, shutting down the reactor. The pressure increases in the steam separator and this pressure increase is maximized by assuming that one steam safety valves and all dump valves to condenser fail to open. The PHOENIX code is used to analyse the transient. Pressure in the steam separator increases to 78 bar. Main cooling pumps rundown provide some circulation and cooling and natural circulation is effective up to 40% full power. In 42 seconds, ECCS flow is established. Adequate fuel cooling is maintained throughout the transient and there are no predicted fuel cladding failures. Cold shutdown with a coolant temperature of 20°C is considered as the final stable state.

III.13.2. Finding

1. The calculated thermal hydraulic behaviour of the primary coolant system during a blackout accident is mild owing to effective natural circulation in the complex RBMK hydraulic circuit.

III.13.3. Plant specific status

Not applicable.

III.13.4. Recommendation

1. Experimental and perhaps commissioning evidence is needed to substantiate the claimed effectiveness of the natural circulation mode of cooling.

Appendix IV
ELECTRIC POWER SUPPLY

IV.1. ELECTRIC POWER REQUIREMENTS

Scope of review: Electric power requirements.

Document/section reviewed: [14] (see Figs 20 and 21)

IV.1.1. Summary of discussions

The principles of main and standby grid connections of the RBMKs were discussed. There are differences between the three generations of RBMKs. The main difference between 1st and 2nd generation is the absence of generator circuit breakers in the connection between generators and main transformers in RBMKs of the 1st generation. At the Leningrad 1 and 2 power plants, a circuit breaker for each generator has already been installed in the course of a backfitting programme. It was stated that similar backfits will be performed on the remaining RBMKs of the 1st generation. The working group supports this plan as it offers an opportunity to supply the in house loads during all operational modes (standby, maintenance, starting up and shutting down phases). In the design of the 2nd and 3rd generation of RBMKs the two generators are connected via one three winding main transformer with the main grid.

The standby power supply design was also discussed. In general there are two standby transformers devoted to two units. Each one of the standby transformers is capable of feeding any of the units by means of a common 6 kV double standby bus.

The main and the standby grid connections at a specific site are tied into one common high voltage switchyard but to sections of different voltages. One concern for the Kursk plant is the fact that one single failure of a high voltage circuit breaker can lead to the loss of one generator and its main grid connection at Unit 2 and one standby transformer of Units 3 and 4.

A lengthy discussion occurred on a major event like the loss of the grid with all units operating and subsequent station rejection and the attempt to maintain in-house power supply for all units by maintaining a stable operation of at least one turbine generator.

The experts discussed the event of the recent fire in Chernobyl 2 and support the intention of RBMK operators and designers to pay very close attention to the operation of the disconnectors and further improvements of the circuit breaker reliability to avoid spurious reconnection of the main generator to the grid.

Under degraded grid frequency conditions (<48.4 Hz) the operator manually reduces reactor power, disconnects both generators from the grid and maintains in-house load.

Relating to transfer from unit transformers to standby transformers in the shutdown from high power, the RBMK designers studied the behaviour of rotating devices as a mini grid of 'generators' and 'loads' (pump motors) during the period of lack of electrical supply. This concept considers essential pumps as loads fed from other equipment running down. This helps to maintain critical flows as high as possible until the equipment is returned to full electrical supply and capacity.

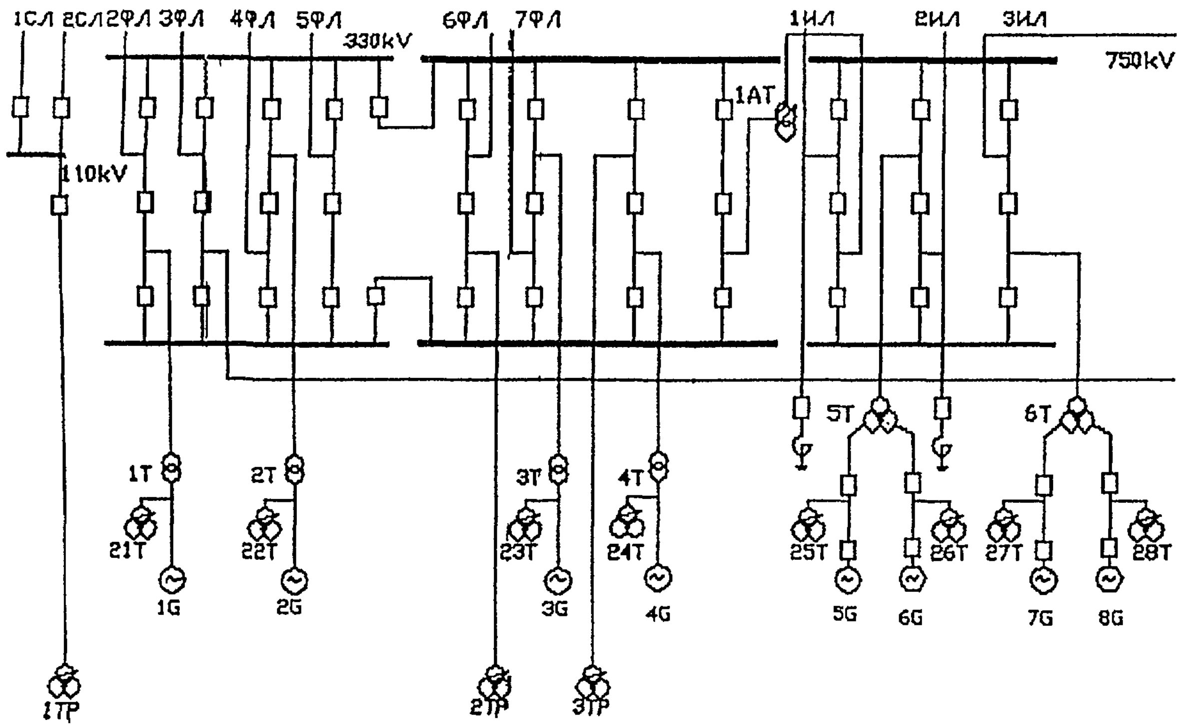


FIG. 20. Main interconnection scheme of Kursk NPP.

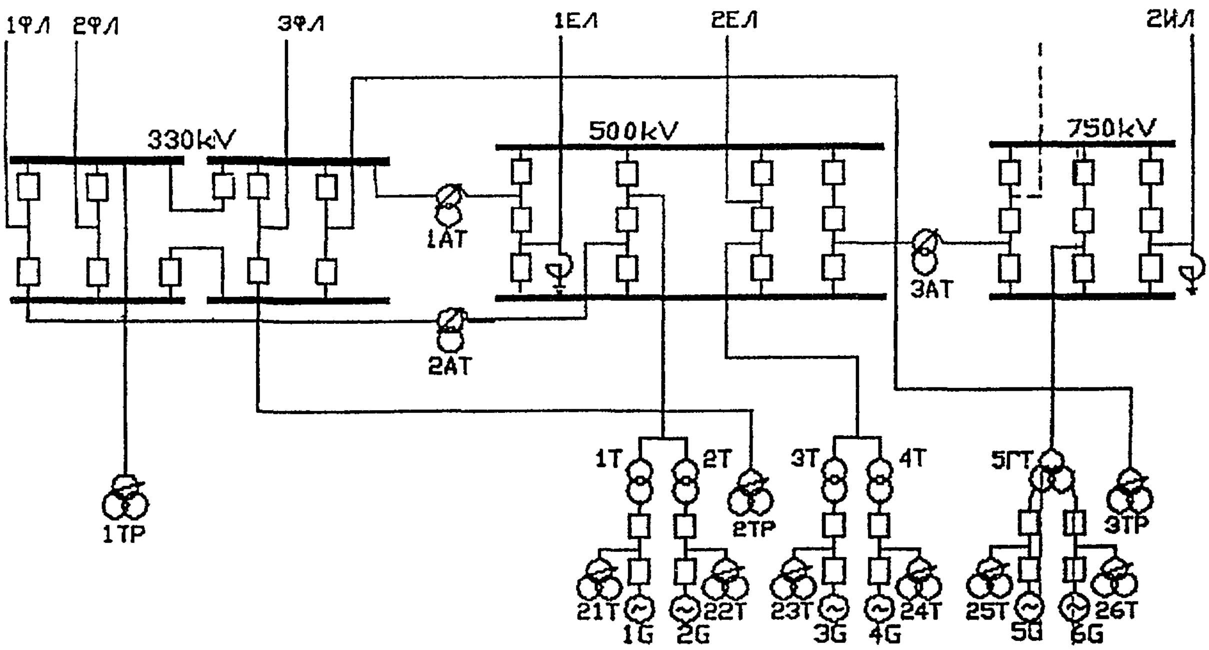


FIG. 21. Main interconnection scheme of Smolensk NPP.

TABLE XVII. ELECTRIC POWER SUPPLY MODIFICATIONS LIST

	Modification	RBMK - 1000			RBMK - 1500	Site
		1st gen.	2nd gen.	3rd gen.		
1	Installation of a circuit breaker between the generator and the main transformer	partially performed	not required	not required	planned (Unit 1)	Done at Leningrad 1 Planned for Unit 2 reconstruction
2	Construction of a building for reliable power supply to safety systems	planned	not required	not required	not required	
3	Construction of diesel generator plant for safety systems	planned	not required	not required	not required	
4	Improvement of protective devices for generators and main transformers	partially performed	planned	not required	not required	Done at Leningrad 1 Planned for Leningrad 2 reconstruction
5	Replacement of generator excitation systems	partially performed	not required	not required	not required	Done at Leningrad 1 Planned for Leningrad 2 reconstruction
6	Substitution of leaktight storage batteries for open ones	planned	planned	not required	not required	
7	Substitution of water for gas for fire extinguishing in cable rooms	planned	not required	not required	not required	
8	Cable coating with fire retardant compounds	performed	performed	performed	performed	
9	Review of the current condition and physical separation of cables running in the same room	planned	partially performed	not required	not required	Done at Kursk 3, 4 and Smolensk 1, 2 By the end of 1992 for the others
10	Partial grounding of 6 kV network neutral	performed	performed	performed	performed	
11	Testing of diesel generators in parallel with the grid during the unit operation	planned	planned	planned	performed	
12	Improvements in reliability of control circuits for high voltage breakers	planned	planned	planned	planned	Following Chernobyl 2 fire
13	Remote control of disconnectors in the generator circuit from the Unit control room	partially performed	planned	planned	planned	Following Chernobyl 2 fire
14	Redundancy for the automatic devices of the grid, involved in disconnection of generators from the grid (System control by generation shedding and load rejection)	performed	performed	performed	performed	

IV.1.2. Findings

1. The first generation of RBMKs is not equipped with generator circuit breakers, but as part of a backfitting program, such breakers have meanwhile been implemented at Leningrad 1 and 2. Within the subsequent generations of RBMKs, however, these generator circuit breakers are already installed.

2. The experts discussed the event of the fire in Chernobyl 2 in October 1991 and were informed that the current design is being modified to remotely control the disconnector in the high voltage switchyard from the unit control room. The operating procedure should also require immediate confirmation of the disconnect opening.

3. All RBMKs are connected to the main grid via main transformers and standby transformers. Examination of the circuit diagram for the Kursk plant revealed a specific situation in which a single failure of one high voltage circuit breaker can lead to the loss of one turbine generator and its main grid connection at Unit 2 and one standby transformer of Units 3 and 4 at the same time (this, however, does not fail a safety function).

4. As a consequence of the Chernobyl 4 accident, the automatic supply of the in-house loads from the main generator isolated from the main grid upon loss of voltage from the grid is now prevented by protective features. The diesel generators maintain power supply to essential loads over an extended period of time until power supply from the grid can be restored.

5. Under degraded frequency conditions however, the operator manually reduces reactor power, disconnects both generators from the grid and maintains in-house load by continuing to operate one of the two generators. Transfer from the unit transformer to a standby transformer is always a 'slow' transfer ($\sim$0.5 s) and special relaying is used to allow critical pumps to continue operation during the transfer.

IV.1.3. Plant specific status

See Table XVII.

IV.1.4. Recommendation

1. An optimization investigation shall be carried out to avoid frequent diesel startup caused by transient voltage fluctuations of the grid. This recommendation falls under the second level of priority.

IV.2. EMERGENCY POWER SUPPLY SYSTEM

Document/section reviewed: [15] (see Figs 22-24)

IV.2.1. Summary of discussions

The general arrangement of safety and safety related channels was explained in detail for the 3rd generation of RBMKs. Three electrically, physically and mechanically separated trains are provided by the design for the safety systems. Each train has 50% of capacity. Additionally there are two safety related channels with 100% capacity each. The two safety related channels are independent from each other. There are a total of 5 diesel generators installed in the 3rd generation

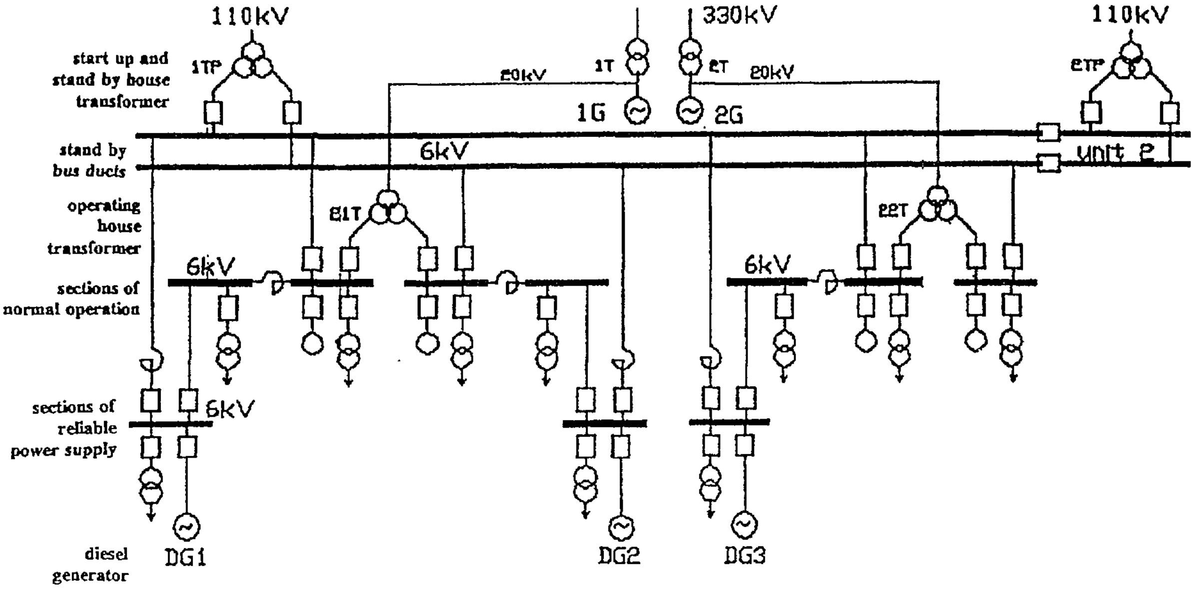


FIG. 22. Scheme of auxiliary power of Leningrad 1 (RBMK 1st generation).

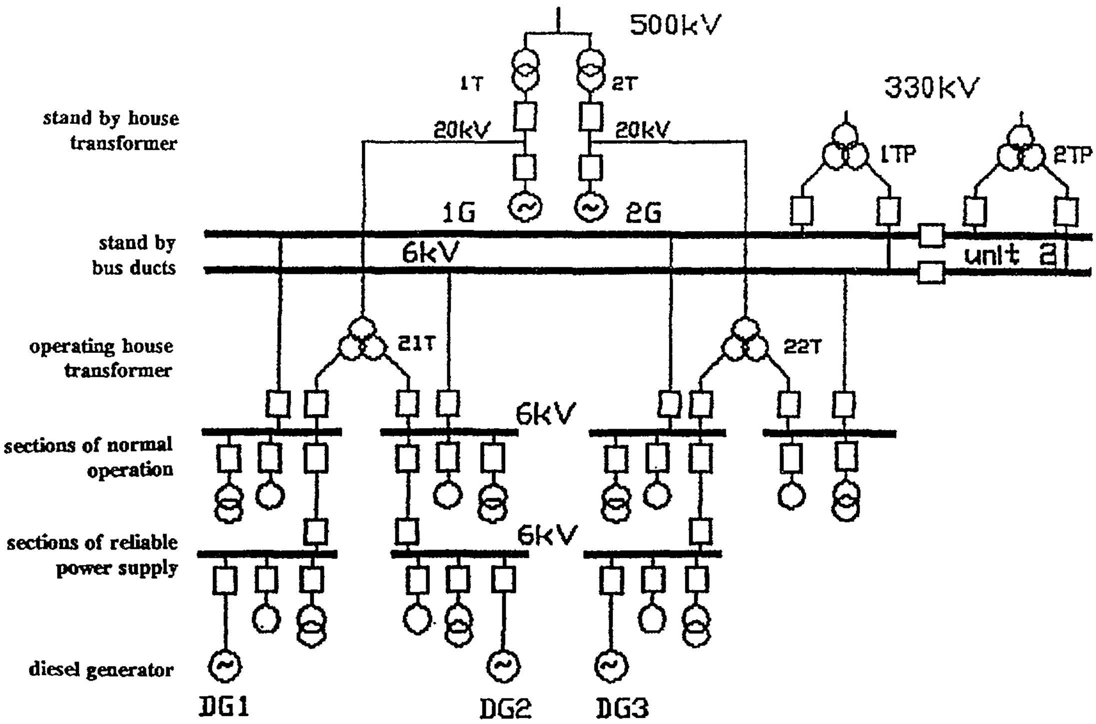


FIG. 23. Scheme of auxiliary power of Smolensk 1 (RBMK 2nd generation).

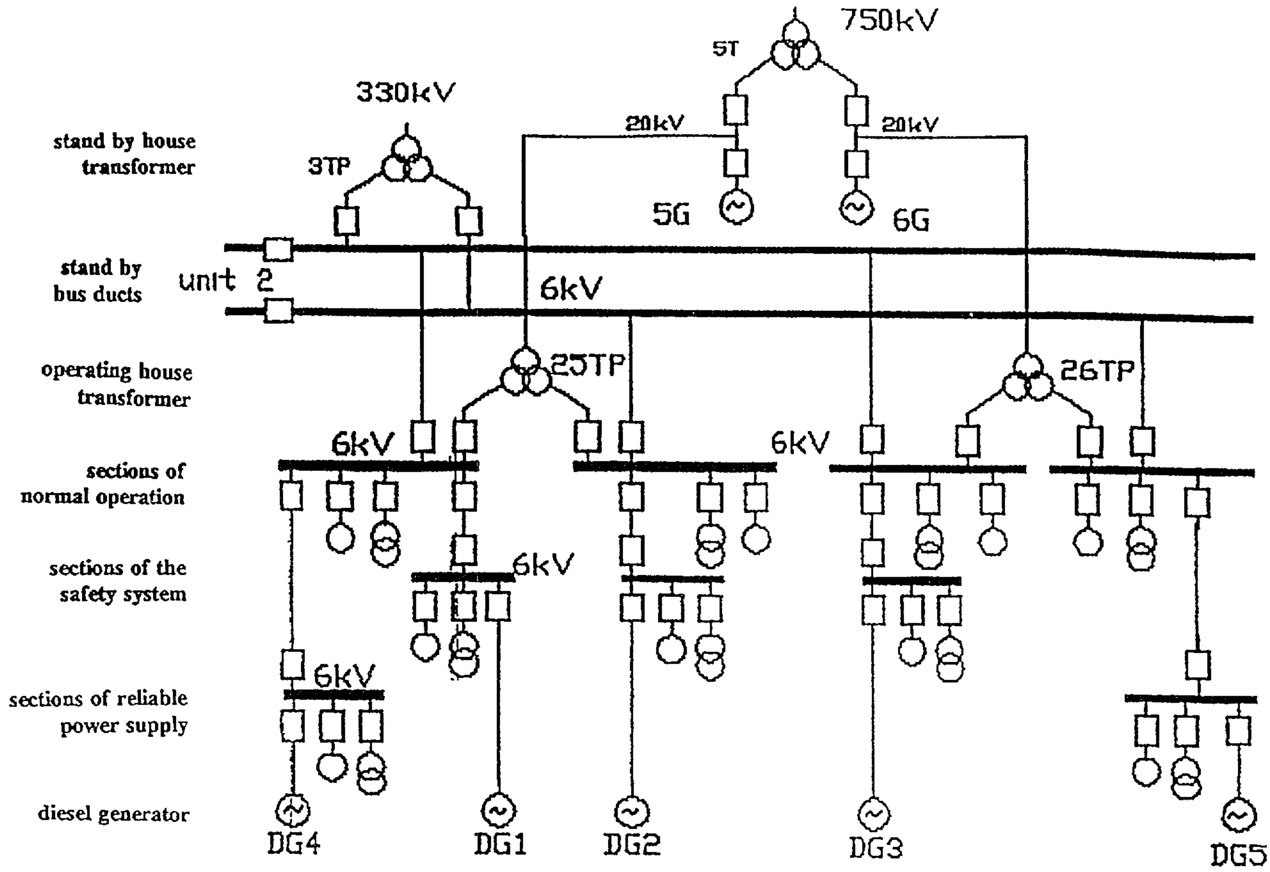


FIG. 24. Scheme of auxiliary power of Smolensk 3 (RBMK 3rd generation).

of RBMKs, three of them are devoted to safety and two of them to the safety related channels. Within the 1st and 2nd generation of RBMKs three diesel generators serve safety and safety related functions at the same time. The philosophy of the design seems to meet the single failure criterion.

Information was provided that strict train separation within the cabling system could not be maintained throughout the plant. Where separation could not be maintained, special features have been used, especially for protection against fire. Further clarification of this aspect will only be possible during site specific inspection.

Due to the procedure of transfer from the unit service transformer to the standby transformer, bus voltage drops for a short duration to the region of 50-25% of the nominal voltage. The diesel generator starts when voltage on the safety bus has decreased down to 25%. It was of concern that there is a possibility (although of low probability) of low bus voltages, but above the diesel generator testing value (see also Section IV.3.2).

The fire retardant cable coating and its ageing, degradation and environmental qualification were discussed and here also the experts have a concern that requires field investigation.

A brief account of the Ignalina systems in this area showed that it is much advanced with 6 diesel generators which can supply the emergency loads in about 35 seconds (ECCS pumps after 20 sec). These are in 6 separate channels with their auxiliaries and loads, and to fulfil all emergency requirements, 4 are needed out of 6.

IV.2.2. Findings

1. Information was provided that strict train segregation within the cabling system could not be maintained throughout the plant. Where separation could not be maintained, special features have been used, especially for protection against fire. Further clarification of this aspect will only be possible during site specific inspection.

2. A backfitting program is planned for the first generation of RBMKs. A new building containing three new diesel generators and their associated support systems to supply the safety buses will be built. The existing three diesel generators are intended to supply normal and safety related systems only.

IV.2.3. Plant specific status

See Table XVII.

IV.2.4. Recommendations

1. Redundant channel separation and protection against common mode failures should be further analysed on a site by site basis; the commencement of this analysis should also be given high priority.

2. Further investigation concerning qualification of electrical equipment under harsh environmental conditions should be performed.

IV.3. DIESEL GENERATORS

Document/section reviewed: [16]

IV.3.1. Summary of discussions

The diesel generators in the emergency power supply system in the second and third generation were discussed. The general conclusion from the review is that these units meet the usual requirements in the experience of the experts with some exceptions already identified in this report and in the course of the discussion. One concern is the idling of diesel generators during monthly testing and there were some questions about raising reliability through redundancy of some elements.

IV.3.2. Findings

1. An automatic startup signal for the diesel generator will be initiated on a vital bus voltage level of 25%.

2. The startup and loading of the safety diesel generators is completed in approximately 50 s, whereas the ECCS pumps are put in operation after 20 s following the ultimate DBA. Information was provided that the discharge time of the ECCS water accumulators is of the order of 60 to 90 s. The experts have a concern with the tight margin between accumulator discharge and water availability from the diesel generator fed ECCS pumps.

3. A backfitting programme is planned for the 1st generation RBMKs: a new building, containing three new diesel generators and their associated support systems to supply the safety buses, will be built. The existing three diesel generators are intended to supply normal and safety related systems only.

IV.3.3. Plant specific status

See Table XVII.

IV.3.4. Recommendations

1. The undervoltage criterion for automatic diesel generator startup should be raised up to about 80% of the nominal voltage.

2. The backfitting programmes for the RBMKs of the 1st generation should be implemented.

IV.4. DIRECT CURRENT SYSTEM

Document/section reviewed: [17] (see Figs 25-27)

IV.4.1. Summary of discussions

The DC circuits of the second and third generation RBMK, together with invertor generated AC, appears to meet all normal requirements. As long as the physical realization is as described, the needs of the reactor units would be met satisfactorily.

There are minor differences between the second and third generation plants.

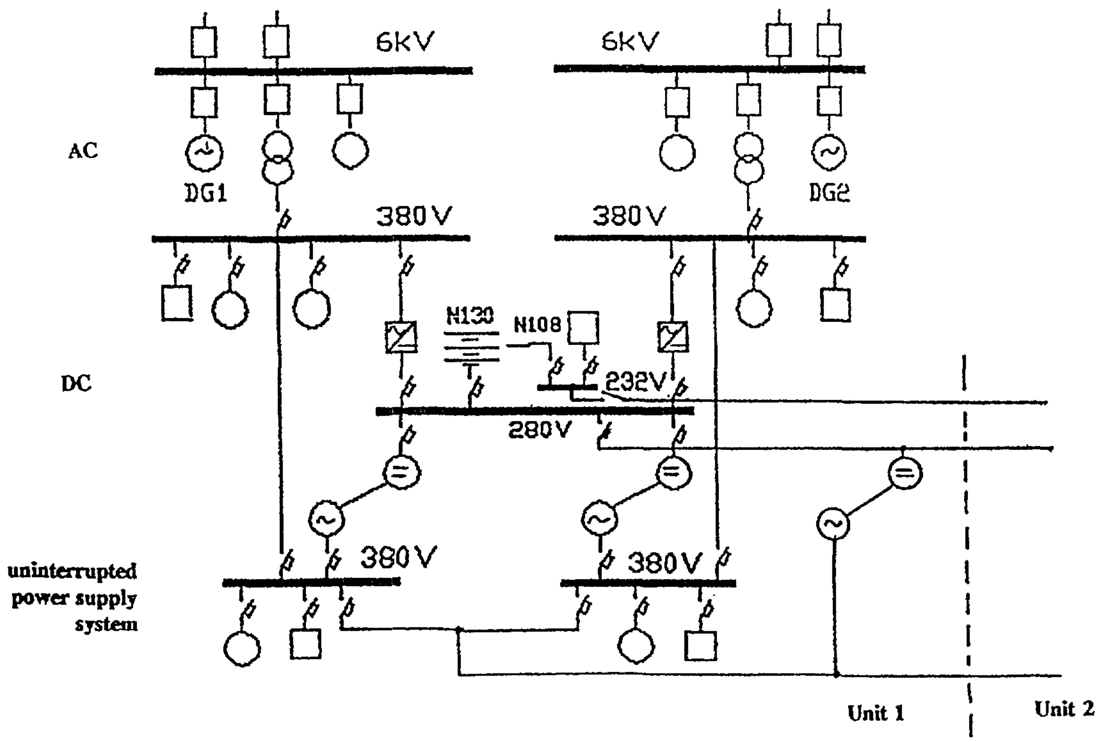


FIG. 25. Scheme of the reliable power supply train of Leningrad 1 (RBMK 1st generation).

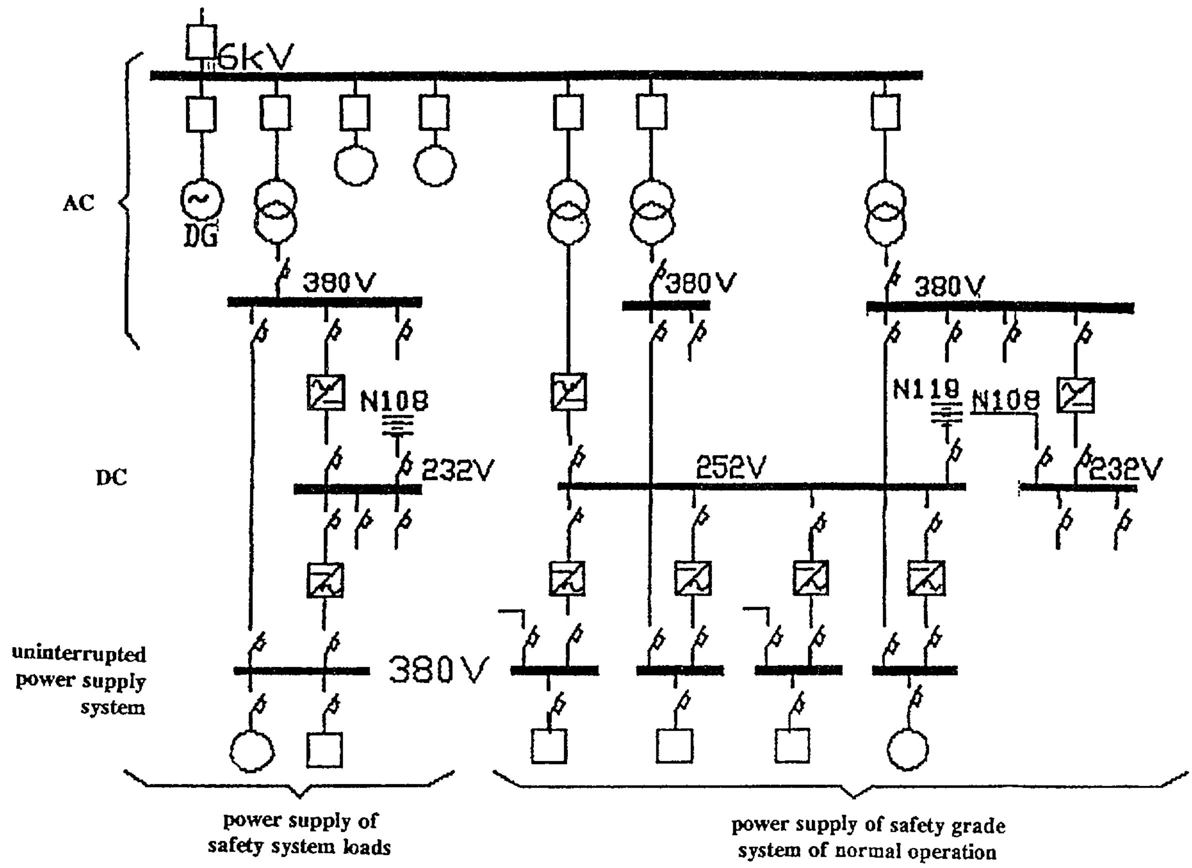


FIG. 26. Scheme of the reliable power supply train of Smolensk 1 (RBMK 2nd generation).

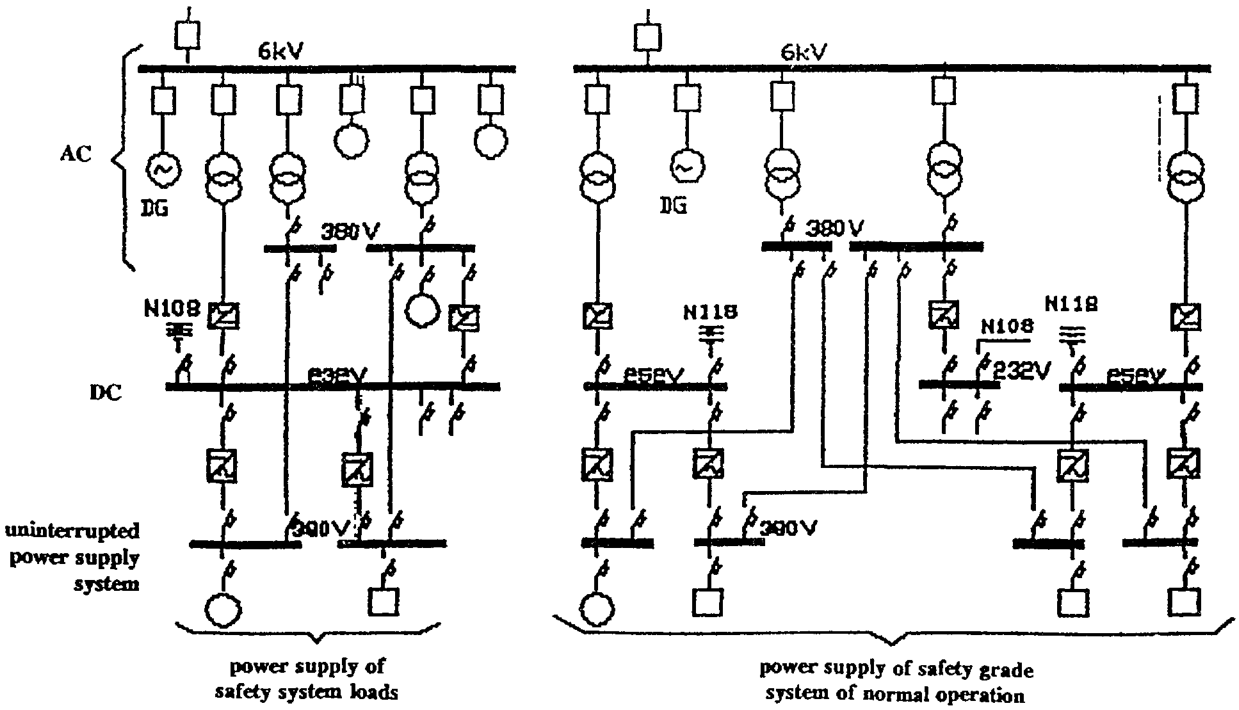


FIG. 27. Scheme of the reliable power supply train of Smolensk 3 (RBMK 3rd generation).

In the first generation RBMK, the DC supply is from a single battery and from a single sectionalized bus. This sectionalized bus supplies all three safety channels. A single failure can disable all safety circuits. In this case there is a possibility to recover DC supply by manual connection of batteries of the other unit. This will, however, take certain time. A change should be planned on high priority to install three DC buses and three batteries. These could be supplied from the three existing alternating current buses. The change should be co-ordinated with the larger backfit plan for new diesel generators and power circuits.

During the discussions of the DC systems, the experts were informed about the designed battery discharge time, which presently is 30 minutes for all the three generations of RBMKs. In the past, this was also the case in plants in western countries. Meanwhile, it became a standard approach to raise the extended battery discharge time to the order of 1 hour or more to be able to cope with station blackout and accident management requirements.

The experience reported on the performance of reversible motor generators did not indicate specific problems.

Additionally, concern was raised about the use of battery cell selector switches in the 1st generation to maintain the DC bus voltage within specified limits. The reliability of operation under accident conditions is the main concern as this device has to perform active actions.

IV.4.2. Findings

1. In the first generation of RBMK plants the DC supply is from a single battery and one single sectionalized bus, feeding three safety channels. One single failure can disable all safety circuits. DC power can only be recovered after a certain time by manual connection to the battery of the other unit.

2. Battery cell selector switches are used within the 1st generation plants to maintain the DC voltage within the specified range.

3. The designed battery discharge time is presently 30 minutes, for all the three generations of RBMKs.

IV.4.3. Plant specific status

See Table XVII.

IV.4.4. Recommendations

1. A change for the first generation of RBMK plants should be planned with a high priority to increase the degree of redundancy of DC buses and batteries. The degree of redundancy of the DC system should be in accordance with the existing 3 channels of the 6 Kv safety supply system. These could be electrically supplied from the three existing alternating current buses. This change could probably be co-ordinated with the large backfit plan for new diesel generators and power circuits, when it is put into effect.

2. When relocating the above mentioned batteries into the new 'emergency power supply building', no battery cell selector switches shall be used for reliability reasons. Train separation shall be maintained from the new equipment in this building down to the safety equipment inside the existing plant.

3. The designed battery discharge time for all RBMK generations currently is 30 minutes. The experts therefore recommend the implementation of batteries of higher capacities in the order of 1 to 2 hours, based on an analysis of system behaviour under station blackout conditions.

REFERENCES

In order to facilitate the review, supporting documentation was prepared by RBMK specialists in English under the co-ordination of RDIPE, Moscow. These documents were not published in the open literature. The areas covered by this documentation are given below.

[1] BORSHCHEV, V.P., PODLAZOV, L.N., Core Design Part 1 Neutron and Physical Design and Operational Calculation, RDIPE, Moscow (1992).

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LIST OF ABBREVIATIONS

AA	absorber assemblies
AC	alternating current
ALS	accident localization system
AR	automatic control rods for controlling power
ASS	additional scram system
ASSET	Assessment of Safety Significant Events Teams
BDBA	beyond design basis accident
BWR	boiling water reactor
CANDU	pressurized heavy water moderated and cooled pressure tube type reactor
COD	crack opening displacement
CPS	control and protection system
CR	control rods
DBA	design basis accident
DC	direct current
DCS	cooling system for damaged parts of the reactor
DEGB	double ended guillotine break
DG	diesel generator
DGH	distribution group header
DHC	delayed hydrogen cracking
ECCS	emergency core cooling system
EFP	emergency feedwater pumps
FA	fuel assemblies
f_{AX}	radial power distribution coefficient
f_R	axial power distribution coefficient
FSS	fast scram system
I & C	instrumentation and control
IE	initiating event
J_R	fracture resistance curve
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LSS	level scram system
LOCA	loss of coolant accident
MCP	main circulation pump
MCR	manual control rods
NDE	non-destructive evaluation
OPB-88	Soviet safety standard issued in 1988
ORM	operating reactivity margin
PSA	probabilistic safety analysis/assessment
PSS	power scram system
PWR	pressurized water reactor
RBMK	boiling water cooled graphite moderated pressure tube type reactor
RDIPE	Research and Development Institute of Power Engineering
SBR	short control rods inserted from the bottom of the reactor
SD	steam separator
SI	safety injection system
SKALA	computerized control and monitoring system
SS-(No)	respective scram system
TITAN	computerized control and monitoring system
UDBA	ultimate design basis accident
USNRC	United States Nuclear Regulatory Commission
VNIIAES	All Union Scientific Research Institute for Nuclear Power Plant Operation
UCS	cooling system for undamaged parts of the reactor
Δk	reactivity worth
β	reactivity equivalent to the delayed neutron fraction

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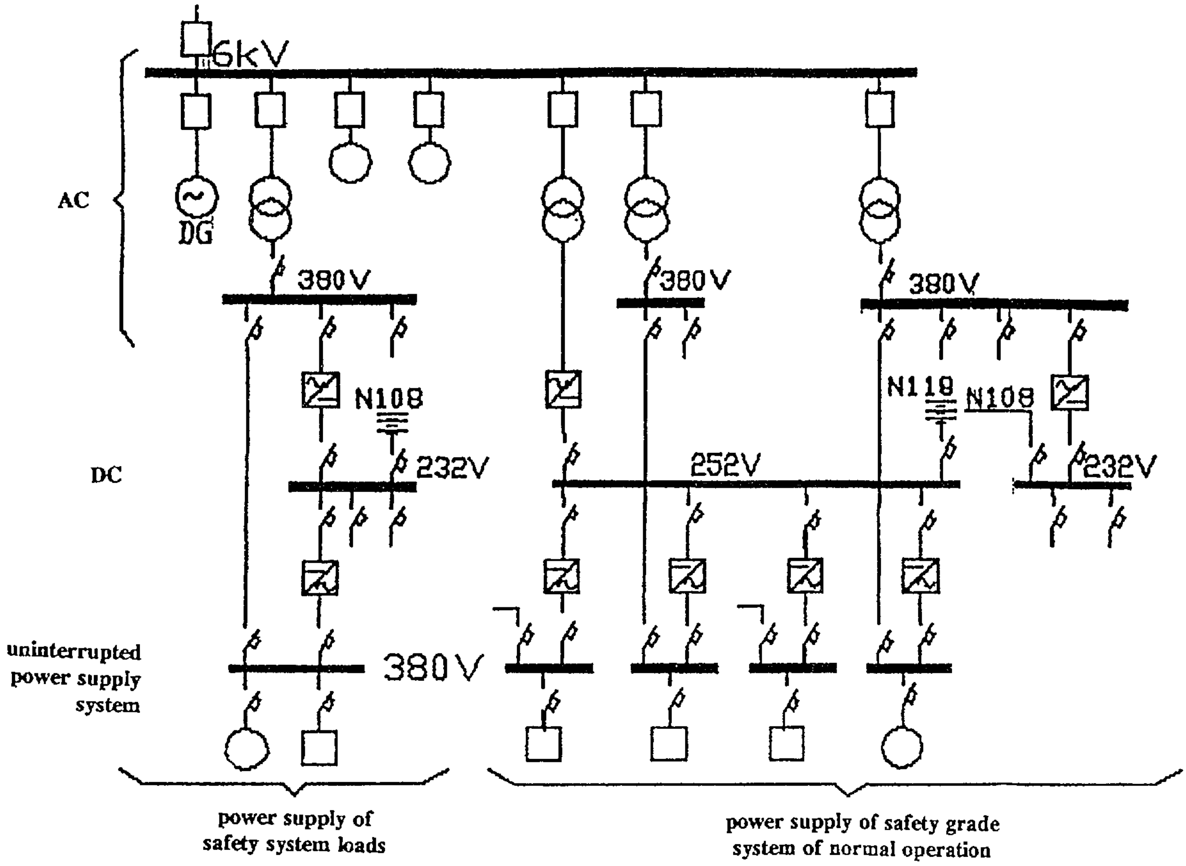

FIG. 26. Scheme of the reliable power supply train of Smolensk 1 (RBMK 2nd generation).

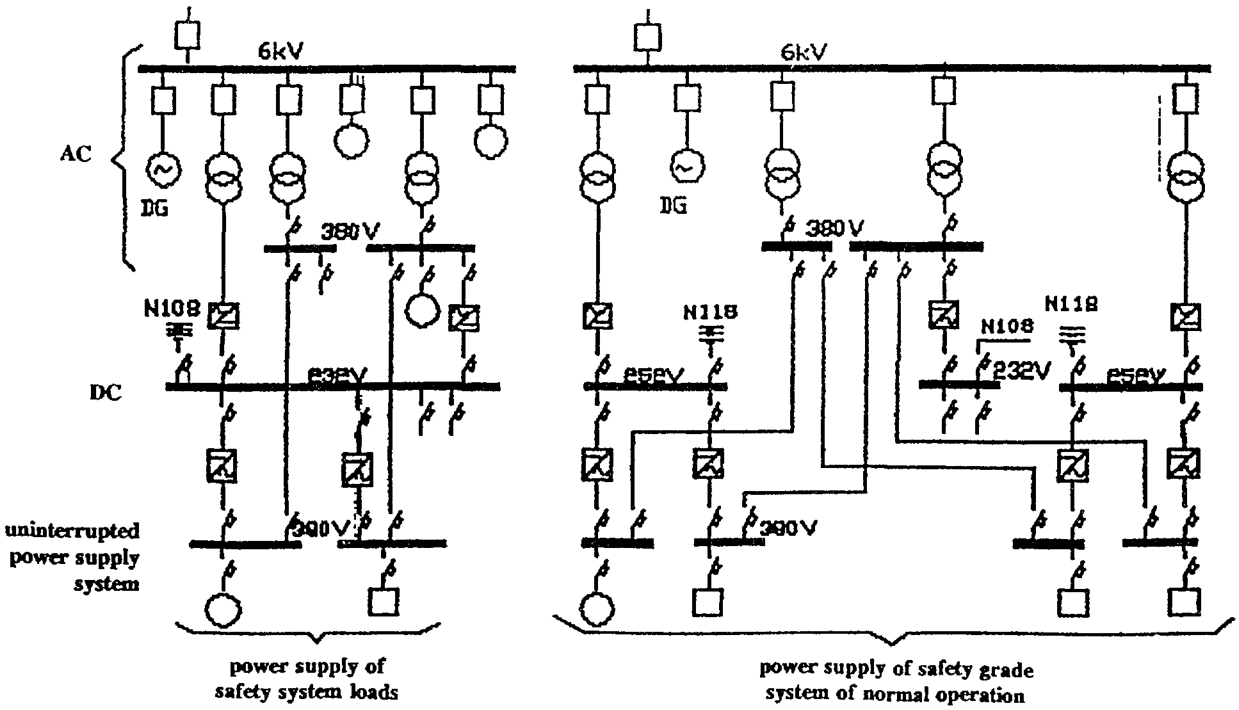

FIG. 27. Scheme of the reliable power supply train of Smolensk 3 (RBMK 3rd generation).

In the first generation RBMK, the DC supply is from a single battery and from a single sectionalized bus. This sectionalized bus supplies all three safety channels. A single failure can disable all safety circuits. In this case there is a possibility to recover DC supply by manual connection of batteries of the other unit. This will, however, take certain time. A change should be planned on high priority to install three DC buses and three batteries. These could be supplied from the three existing alternating current buses. The change should be co-ordinated with the larger backfit plan for new diesel generators and power circuits.

During the discussions of the DC systems, the experts were informed about the designed battery discharge time, which presently is 30 minutes for all the three generations of RBMKs. In the past, this was also the case in plants in western countries. Meanwhile, it became a standard approach to raise the extended battery discharge time to the order of 1 hour or more to be able to cope with station blackout and accident management requirements.

The experience reported on the performance of reversible motor generators did not indicate specific problems.

Additionally, concern was raised about the use of battery cell selector switches in the 1st generation to maintain the DC bus voltage within specified limits. The reliability of operation under accident conditions is the main concern as this device has to perform active actions.

IV.4.2. Findings

1. In the first generation of RBMK plants the DC supply is from a single battery and one single sectionalized bus, feeding three safety channels. One single failure can disable all safety circuits. DC power can only be recovered after a certain time by manual connection to the battery of the other unit.

2. Battery cell selector switches are used within the 1st generation plants to maintain the DC voltage within the specified range.

3. The designed battery discharge time is presently 30 minutes, for all the three generations of RBMKs.

IV.4.3. Plant specific status

See Table XVII.

IV.4.4. Recommendations

1. A change for the first generation of RBMK plants should be planned with a high priority to increase the degree of redundancy of DC buses and batteries. The degree of redundancy of the DC system should be in accordance with the existing 3 channels of the 6 Kv safety supply system. These could be electrically supplied from the three existing alternating current buses. This change could probably be co-ordinated with the large backfit plan for new diesel generators and power circuits, when it is put into effect.

2. When relocating the above mentioned batteries into the new 'emergency power supply building', no battery cell selector switches shall be used for reliability reasons. Train separation shall be maintained from the new equipment in this building down to the safety equipment inside the existing plant.

3. The designed battery discharge time for all RBMK generations currently is 30 minutes. The experts therefore recommend the implementation of batteries of higher capacities in the order of 1 to 2 hours, based on an analysis of system behaviour under station blackout conditions.

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